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ECL 216

Design of a FAST REACTIVITY EXCURSION DEVICE

This case history concerns the design, development, and testing of the Fast Reactivity Excursion Device (FRED) by the Nuclear Energy Division of the General Electric Company, San Jose. This is a device which was to remove a nuclear poison rod from the core of an experimental breeder reactor in less than 0.1 second. The experimental reactor was built to demonstrate the self-limiting characteristic of liquid-metal-cooled fast breeder reactors. The FRED was required to simulate a sudden increase in the reactor power. Thus, the inherent safety characteristic of automatic shut-down in a hypothetical accident would be demonstrated. The FRED played a central role in the main objective for constructing this experimental reactor, and it attracted much attention from all parties concerned. The case examines how one engineer successfully uses his own design philosophy to develop a reliable device, and how the results of testing the device as well as changes in the functional requirements during the development affected the final design.

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1. INTRODUCTION

Although the first nuclear reactor used for electric power generation was an experimental breeder reactor,[1] there were virtually no breeder reactors in use for commercial power generation in the United States at the time the project presented in this case was undertaken. The main reason for the lag in the use of breeder reactors instead of conventional water, or "thermal" reactors[2] lies in the slow development of a suitable breeder reactor fuel.

Early breeder reactors used fissionable uranium fuel in the metallic form (see Table 1.1) because that type of fuel exhibited a very strong and fast-acting negative power coefficient. This was a very important property, since the inherent safety of a reactor depends on its negative power coefficient.

1.2 The Power Coefficient

The power coefficient describes the reactor core response to changes in the control inputs. A reactor with a net negative power coefficient would not increase in power as much as the control rod movement implied. If a reactor had a net positive power coefficient, the power increase determined by the control rod movement would be amplified, and possibly cause the reactor to go out of control. A reactor with a strong negative power coefficient will shut itself down to a safe level if the control rods were suddenly removed, as in a hypothetical situation. Conventional thermal reactors have a characteristic strong negative power coefficient because the moderator slows down fewer neutrons when it begins to boil, and therefore, is less capable of maintaining the required chain reaction.

In breeder reactors the power coefficient is a much more important characteristic, since the neutrons are used at the speed at which they are emitted, approximately 10^8 times faster than in thermal eeactors. The strong negative coefficient of the early breeder reactors was due to the thermal expansion of the metallic fuel, which caused a reduction in the number of neutrons captured by the fertile material in the fuel.[3] Unfortunately, the metallic uranium fuel deteriorated under irradiation and required frequent changes, making it undesirable for use in commercial power plants.

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[1]EBR-I (Experimental Breeder Reactor), December, 1951.

[2]For an explanation of thermal and breeder reactor theory, see Appendix A.

[3]The fertile atoms convert into fissionable atoms upon capturing a neutron, and thus maintain the nuclear reaction in the reactor core.

TABLE 1.1

Early Fast Breeder Reactors

	U.S.A.				USSR			UK	FRANCE
	Clementine	EBR-1	EBR-11	Enrico Fermi	BR-1	BR-2	BR-5	Dounreay	RAPSODIE
Reactor Power									
Thermal MW	0.025	1.2	62.5	430	0	0.1	5	72	20
Electrical MW	0	0.2	20	150	0	0	0	15	0
Fuel	Plutonium Metal	Uranium Metal	Uranium Metal	Uranium Metal	Plutonium Metal	Plutonium Metal	Plutonium Oxide and Plutonium Carbide	Uranium Metal	Uranium and Plutonium Oxide
Coolant	Mercury	Sodium-Potassium	Sodium	Sodium	--	Mercury	Sodium	Sodium-Potassium	Sodium
Time Schedule									
Design	1945	1945	1956	1956			1956	~1954	1958
Operation	1949	1951	1965	1966	1955	1956	1959	1963	1967
Shutdown	1953	1963	--	--	1956	1957	?	--	--
Reactor Incidents	--	Core Meltdown	None	Partial Core Meltdown	--	--	--	Leak in Primary Coolant System	None

1.3 Incentives for Developing Breeder Reactors

Thermal reactors gained widespread use throughout the world over the years, but there were still strong incentives for developing a breeder reactor that had a long lasting fuel. Conventional reactors utilize only about 1% of the energy content of the uranium fuel. Breeder reactors, by converting abundant amounts of fertile U-238 into fissionable Plutonium, utilize about 70% of the energy content of the fuel. A. S. Gibson of General Electric, in "The Fast Breeder Reactor", estimates that conventional reactors could produce only 1 or 2 Q of energy from our uranium resources, where $1Q = 10^{18}$ Btu. Fossil fuel reserves costing no more than four times the present costs equal about 30Q of energy and could last about a century, with severe shortages before the end of that time. Breeder reactors could extract an estimated 1000 Q or more of useful energy, thereby extending the use of uranium over several centuries. Thus, one can see the strong motivation to develop breeders.

In addition, there are ecological and economic incentives for the development of a practical breeder reactor. Breeder reactor power plants can operate at a higher thermal efficiency than either fossil-fueled or conventional water reactor power plants. They expel significantly less gaseous, liquid and solid wastes,[4] and can produce electric power more economically. A General Electric study determined that the savings due to the reduced cost of generating electrical power would be more than $45 billion between the time the breeder reactors could be commercially introduced in the 1980's and the end of this century.[5]

1.4 Developments Leading to SEFOR

In the late 1950's, General Electric began a program to develop a long lasting breeder fuel. It was found that a ceramic type fuel could withstand higher operating temperatures and was long lasting. In 1957, G. E. proposed to the U. S. Atomic Energy Commission a program to develop a long-lived breeder fuel. Work was started on the test and development program in 1959. But, because of fabrication techniques, the thermal expansion coefficient of the proposed ceramic fuel was less predictable than that of the metal fuel. If ceramic fuel were to be used in breeder reactors, it had to be characterized by some type of strong and prompt negative power coefficient.

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[4] For a comparison of the wastes produced by equal sized coal-fired plant, a water reactor plant and a breeder reactor plant, see Appendix B.

[5] "Incentives for the Development of the Fast Breeder Reactor, P. M. Murphy.

In 1959, Dr. Paul Greeber of G. E. and Dr. R. Nicholson independently found that a ceramic oxide fuel did possess a strong, prompt negative power coefficient which was as predictable as the expansion coefficient of the metallic fuel. This negative coefficient depended on what is known as the Doppler effect, in which the uranium atoms reach a "resonant" energy level when they heat up, and capture more neutrons without fissioning. This action takes place at the same rate as the rate of increase in the available neutrons, and therefore, the Doppler effect occurs fast enough to control the power surges in a reactor, and would shut down the reactor to a safe level.

In 1961, G. E. proposed the construction of a small test reactor to study the Doppler effect in operation. In 1962, the Southwest Atomic Energy Association (SAEA)[6] requested proposals concerning the development of long range, low cost nuclear power. General Electric responded to this request, along with the German Karlsruhe Laboratory, with assistance to be given by the U. S. Atomic Energy Commission.

2. THE SEFOR PROGRAM

These four organizations launched the Southwest Experimental Fast Oxide Reactor (SEFOR) program in March 1964. The main objectives of the SEFOR program were to study the Doppler effect in action, to measure the Doppler coefficient of the reactivity, and to demonstrate the ability of the negative Doppler coefficient to shut down the reactor in case of a sudden power increase.

In the SEFOR program, SAEA wanted no obligations for day to day technical work. Karlsruhe, which was receiving financial assistance from the government of West Germany and Euratom, wanted to contribute to the design, development and construction activities. The AEC wanted to be kept informed on all matters, and to participate only in specific areas. General Electric had the responsibility for performing work under general guidelines established by committees of the four organizations.

Funds for the design and construction of SEFOR were provided by the four participating organizations. AEC provided about about $12 million for the research and development programs. SAEA contributed about $10 million, mostly in construction cost. The Karlsruhe Laboratory contributed about $3 million. General Electric invested a substantial amount in their facilities and contributed about $1.5 million, and was to absorb any cost overruns.

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[6]A group of seventeen investor-owned utility companies of the South and Southwestern United States.

2.1 General Electric Company Goals

There were three company goals which General Electric wished to obtain by their participation. G. E. recognized that commercial development of the breeder reactor would be an immense undertaking. Therefore, they felt that technical information from SEFOR was necessary before they embarked on any future breeder reactor plants. Secondly, General Electric wanted to make use of the organization established for the SEFOR program to carry out the major steps ahead in the fast breeder field. And finally, General Electric desired to have first-hand experience on the design, construction and operation of a liquid-metal-cooled fast breeder reactor, built on a commercial basis, to determine if the assessment of the sodium cooled fast breeder reactor was justified. These goals were established to help G. E. prepare for possible participation in the Demonstration Plant Project[7] to be built in 1980.

2.2 General Electric SEFOR Organization

The Nuclear Energy Division of General Electric was responsible for the SEFOR project. The research and development program was divided into ten tasks as shown in Table 2.1. The organization of the design, construction and operation of the SEFOR project in 1964 is shown in Fig. 2.1 Plant managers and project engineers from various departments and organizations met in Technical Policy Meetings about every six months to review the progress of the ten tasks. The SEFOR Safety Review Committee provided audits of various aspects of the project and reported to the manager of the Advanced Product Operations. The SEFOR Site Safety Committee reviewed the procedures and methods for accomplishing the experimental programs to assure safe operation of the facility.

3. FAST REACTIVITY EXCURSION DEVICE

In the analytical study of the Doppler effect, Dr. Paul Greebler was able to calculate the Doppler coefficient to 11 figure accuracy by using a computer model. However, General Electric strongly felt that his predictions should be confirmed under actual reactor operation conditions.

The Fast Reactivity Excursion Device (FRED) was developed to initiate a reactor power excursion in order to simulate a very rapid rise in the reactor power level. These excursions were

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[7]A liquid-metal fast breeder reactor (LMFBR), capacity of 300-500 MW, using liquid sodium as a coolant; the prime contract was to be awarded late in 1972.

Table 2.1

R & D Program

TEN TASKS

Task 1: Planning the experimental program
 Task 1.1: Specifications for the reactor and experimental program

Task 2: Experimental program in ZPR III (Zero Power Reactor)

Task 3: Fuel Development
 Task 3.1: Fuel clad transient performance
 Task 3.2: Specification of fuel composition

Task 4: Special Instrumentation
 Task 4.1: Fuel temperature measurement
 Task 4.2: Core flux detector system
 Task 4.3: Failed fuel locator
 Task 4.4: Instrument lead connectors
 Task 4.5: Other core instrumentation - Flow meter, Thermocouples

Task 5: Steady state and transient characteristics of the core assembly
 Task 5.1: Core mock-up and heat fuel test
 Task 5.2: Hydraulic flow tests (vessel)
 Task 5.3: Hydraulic flow tests (fuel rods)
 Task 5.4: Sodium performance tests
 Task 5.5: Thermal shock tests
 Task 5.6: Sodium heat-transfer tests

Task 6: Reactivity controls
 Task 6.1: Fast reactivity excursion device (FRED)
 Task 6.2: Oscillator drive

Task 7: Fuel handling
 Task 7.1: Refueling cell mock-up
 Task 7.2: Test of remote handling procedures
 Task 7.3: Fuel inspection
 Task 7.4: Sodium vapor control.

Task 8: Core design
 Task 8.1: Nuclear analysis
 Task 8.2: Mechanical design

Task 9: Fuel fabrication

Task 10: Plant construction

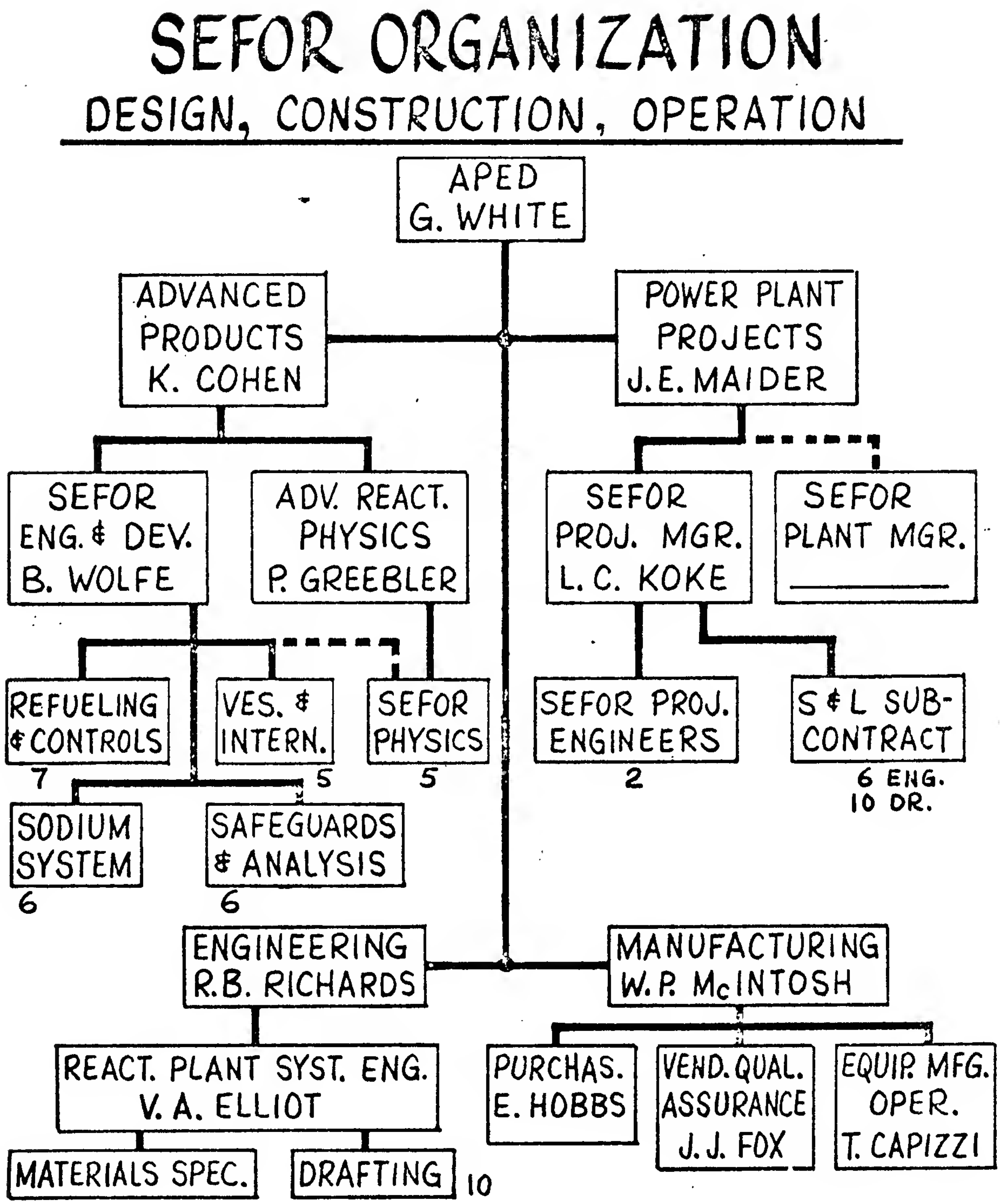

FIGURE 2.1 - SEFOR Organization

achieved by rapidly removing a poison[8] slug from the reactor core. This device was used to carry out the transient testing phase of the SEFOR experimental program, demonstrating that these excursions were accompanied by a strong negative reactivity. Thus the predictability of the Doppler coefficient and the inherent safety characteristic of the breeder reactor in a hypothetical accident was verified.

The FRED was to be supported on the reactor vessel head shown in Fig. 3.1. The position of the FRED at the center position of the reactor vessel is shown in Fig. 3.2.

The design of the FRED was the responsibility of Mr. Ernest R. McKeehan, who was a member of the Refueling and Controls group shown in Fig. 2.1. Several program engineers and technicians were later assigned to assist him. Mr. McKeehan has worked with General Electric for twelve years. He graduated with a B. S. degree in Mechanical Engineering from Oregon State University in 1960, and completed the extensive three year ABC courses[9] at General Electric from 1960 to 1963. He also received the M.B.A. degree from the University of Santa Clara.

3.1 Design Restrictions and Requirements

In November 1964, the FRED functional requirements and design restrictions were issued by the Advance Reactor Physics group (see Fig. 2.1). The following listing of the original requirements and restrictions were established by the G. E. report entitled "Task 6-1, Fast Reactivity Excursion Device Status Report and Preliminary Design", dated November 19, 1964.

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[8] Material of high absorption cross section that absorbs neutrons unproductively and reduces the reactivity of a reactor.

[9] At General Electric, newly recruited engineers may participate in a one year training program. They are given rotating work assignments and can take a three course series (Courses A, B, and C) in advanced engineering analysis.

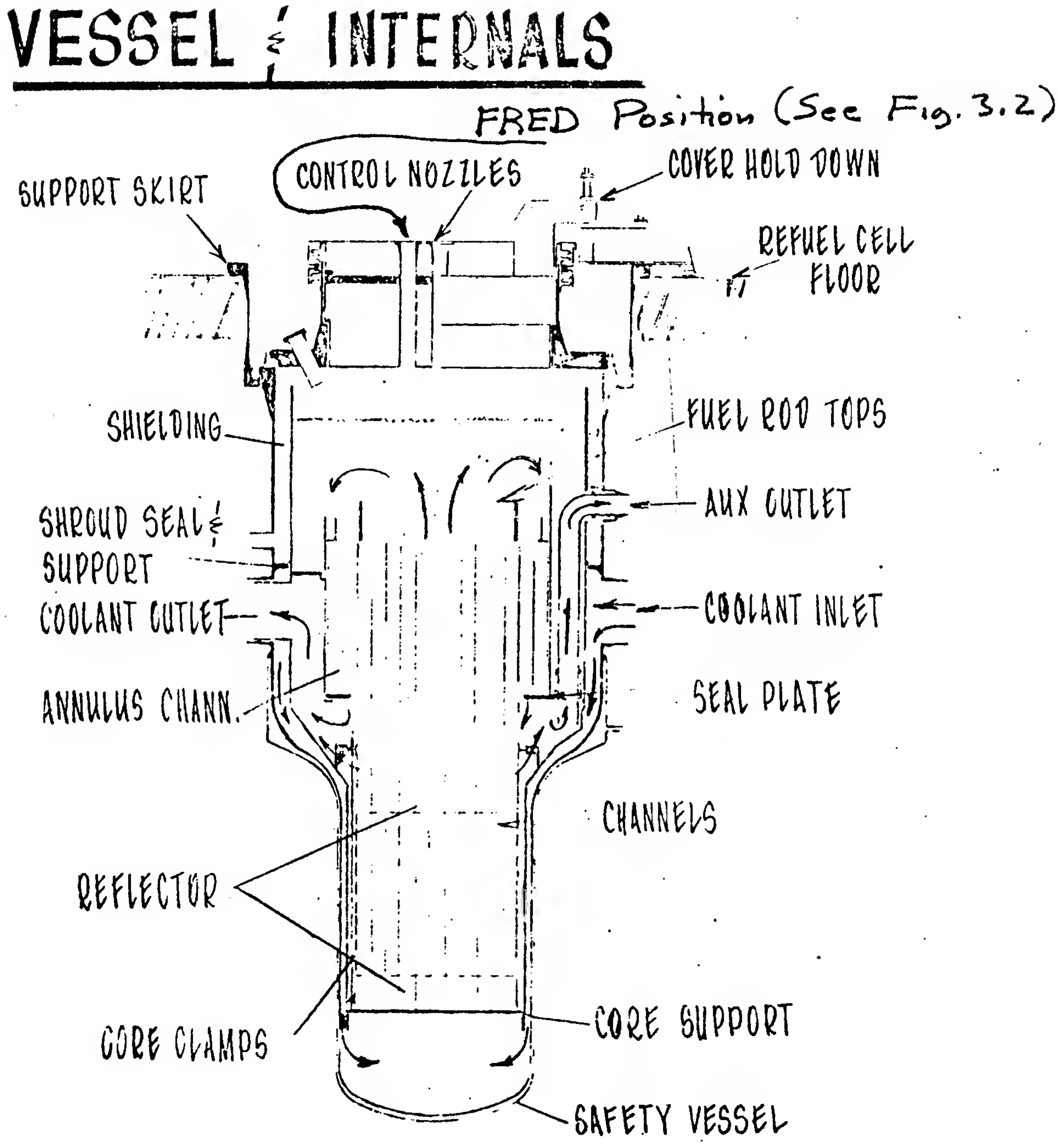


FIGURE 3.1 - FRED Position in Vessel

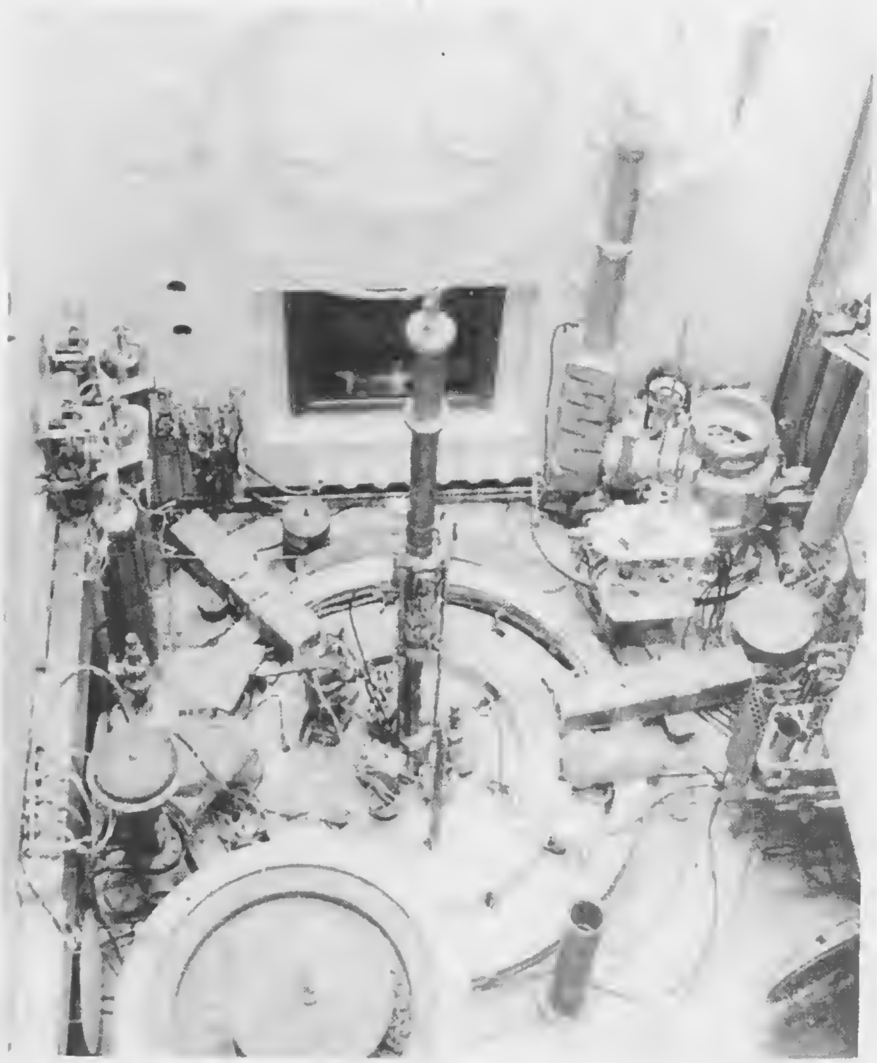


FIGURE 3.2 - Position of FRED on the Reactor Vessel

FUNCTIONAL REQUIREMENTS

1. Removal time 0.050 sec required, 0.025 sec desirable. This time is from initiation of signal at control panel to receipt of signal that slug has left the core.

2. Travel in core - 1/2 core height; 17 inches.

3. Starting Position - constant; center of core.

4. Position Indication - start, midpoint and exit from core - required continuous indication - desirable.

5. Worth - variable in discrete steps, 10¢ to $3.

6. Reactivity vs Position - a small drive will be incorporated to allow for measuring reactivity vs position of a poison slug when it is inserted in core.

DESIGN RESTRICTIONS

Space Occupied:	In core - all of center channel In cell - 7 inch diameter by 12 feet (approximate)
Maximum Reaction Load:	On head - 7000 pounds If greater than 7000 pounds, structure to be supported on floor.
Gas Exhaust:	Should be a gas that will not contaminate cell or it will have to exhaust in a closed system.
Handling:	Installation in head by hand. Remove remotely from head. Not attached to head when head is removed.

The worth referred to in requirement number 5 is the reactivity value of the poison slug that is attached to the FRED. In the fission of U-235, about 99.25% of the neutrons are expelled from the nucleus in about 10^{-14} second, practically instantaneously after fission takes place. These are called prompt neutrons. The remaining delayed neutrons, about 0.75%, leave the fission nuclei fragment over a period of minutes. No control system can handle the short time factor of the prompt neutrons. It is the delayed neutrons that make it possible to control the chain reaction of a nuclear reactor. The term "one dollar's worth of reactivity" has been assigned as being equal to the reactivity due to the delayed neutrons alone. Decimal fractions of dollars worth of reactivity are expressed in cents. Thus, a reactivity less than one dollar is sub-critical (power decreasing). If the reactivity is equal to one dollar, it is critical (power constant). If the reactivity is greater than one dollar, it is super critical (power increasing) or super-prompt critical (power increasing rapidly).

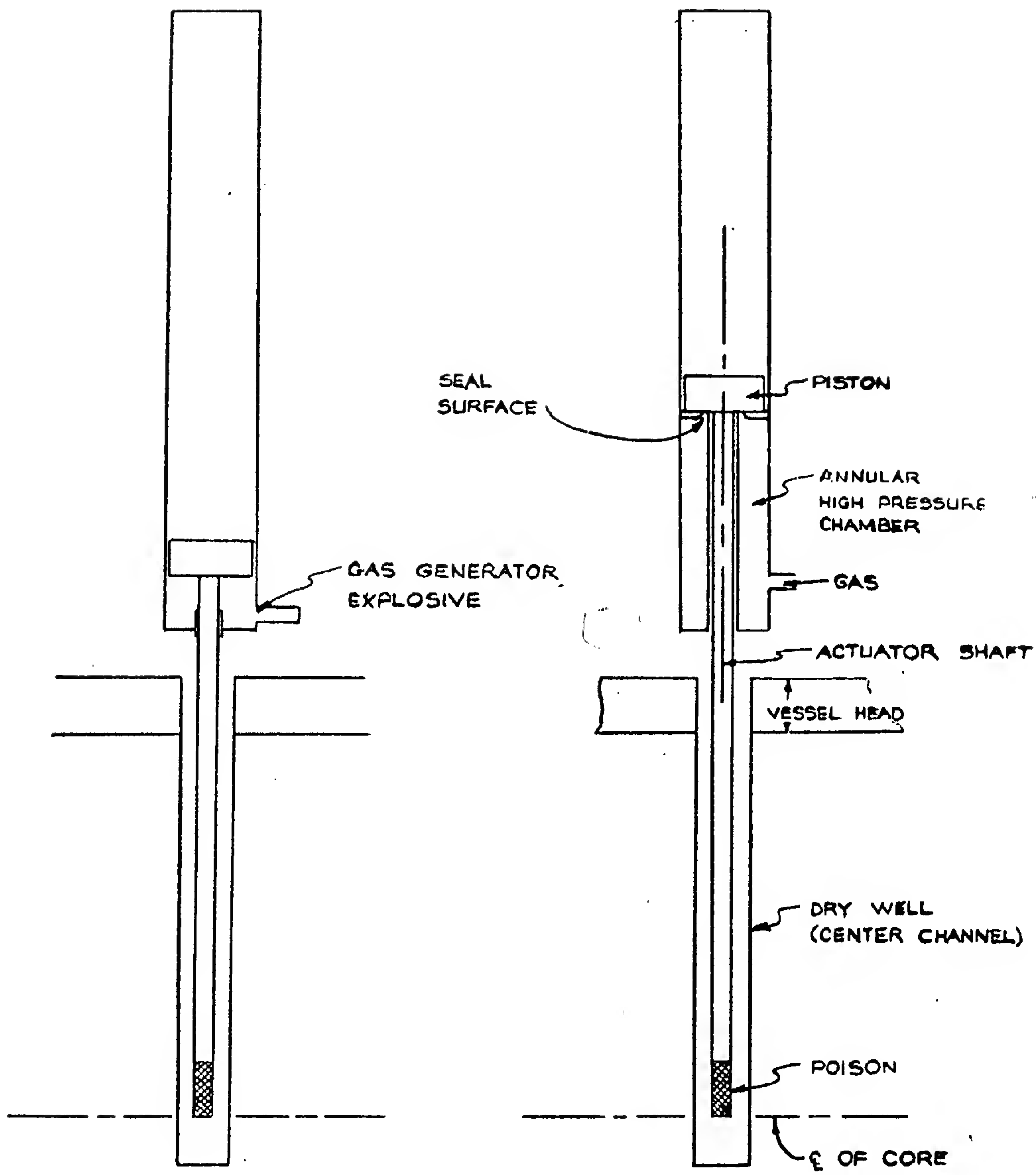
3.2 Proposed Methods

Four methods were proposed for accomplishing these requirements within the design restrictions. Schematics of these methods are shown in Figs. 3.3 and 3.4.

Method 1. A gas mechanism consisting of a gas cylinder, piston, and actuator shaft. This mechanism would be operated by releasing the pressure in the gas accumulator under the piston-actuator shaft. This device uses compressed gas as the propellant. The piston of this device rests on an annular seal surface which exposes a small area on the lower side of the piston to the high pressure propellant gas. When the propellant gas pressure acting on the exposed annular area is great enough to lift the piston, the piston will move up and thus expose all of the lower surface of the piston to the high pressure propellant. The actuator shaft moves up through the center of the annular high pressure chamber. The lower end of the actuator shaft extends inside a dry well down to the center of the core where the poison slug is attached.

Method 2. An explosive device similar to the gas mechanism of Method 1 except that the pressure pulse if furnished by an explosive charge instead of a pressure accumulator. The ignition of the gas generator produces a high pressure gas supply that acts over the lower surface of the piston. After ignition of the gas generator, the action of this device would be similar to Method 1.

Method 3. A momentum transfer device operating on the principle of the impact of a small mass accelerating a larger mass



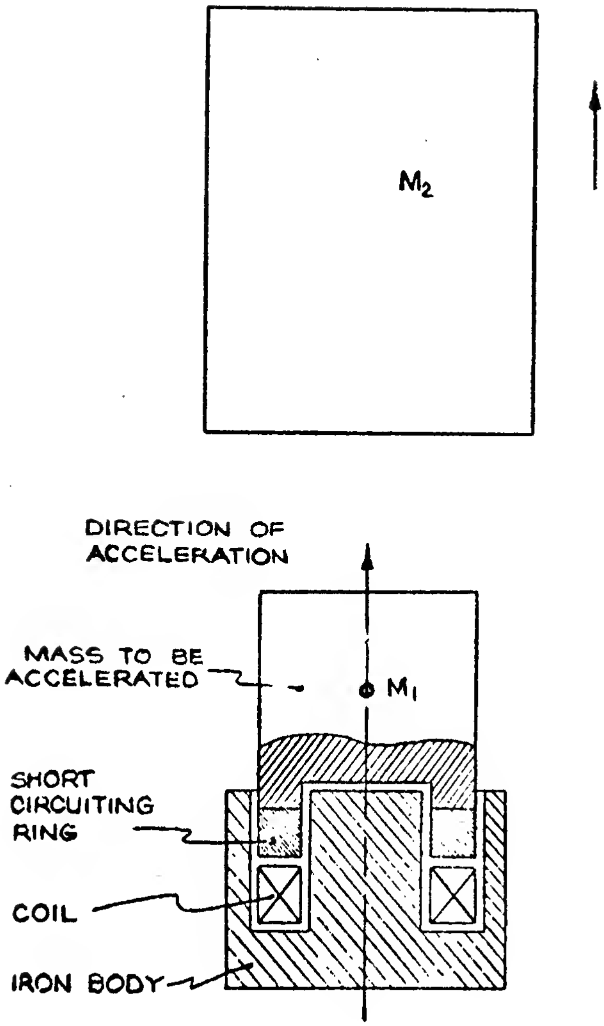
METHOD 2

Gas Generator Device

METHOD 1

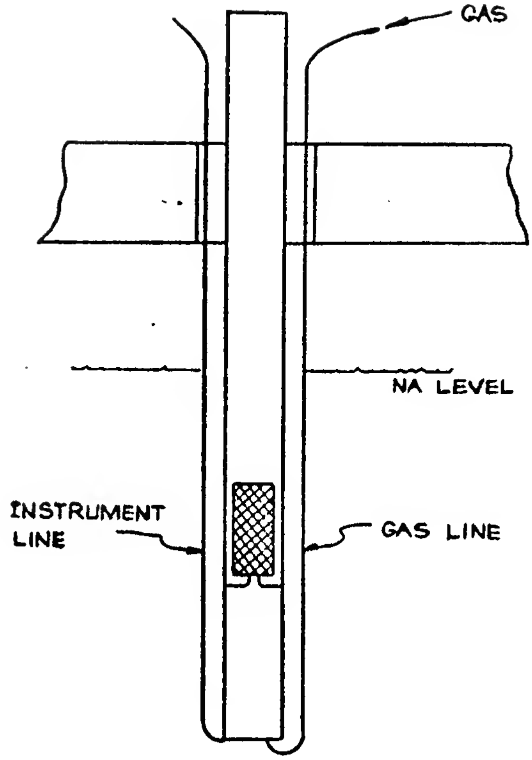
Expanding Pressure Area Outside of Core

FIGURE 3.3 - Methods 1 and 2



METHOD 3

Momentum Transfer Device



METHOD 4

Expanding Pressure Area In-Core

FIGURE 3.4 - Methods 3 and 4

which may be repeated with a series of increasing masses. A device of this type has been designed by the Karlsruhe Laboratory in Germany. The principle of operation was described in G. E. Quarterly Report, dated January 1965.

In this device, the small mass (m_1) is accelerated by the repulsion coil and impacts the larger mass (m_2).

Momentum before impact:

$$M_i = m\ v$$

Momentum after impact-assuming elastic impact (e = 1)

$$M_0 = m_2 v_2 + m_1 v_1$$

Note: v_2 and v_1 are in opposite directions

Then by conservation of momentum:

$$M_0 = M_i$$

$$m_1 v = m_2 v_2 - m_1 v_1$$

Therefore

$$m_2 v_2 = m_1\ (v + v_1)$$

Thus, the second mass has a larger momentum than the small mass had before impact.

With further reductions and algebraic manipulations, the following velocity relations are found for a mass ratio of 1/10.

Velocity of small mass before impact

$$v = 5.5\ v_2$$

Thus, for v_2 of 750 inch/sec (17 in/0.020 sec)

$$v = 4120 \text{ inch/sec}$$

Method 4. The gas-operated mechanism previously tested under the Fast Ceramic Reactor Development Program. It uses the same concept of expanding pressure area as used in (1) above, except this device does not have the actuator shaft to penetrate the high pressure chamber.

These four methods were then carefully evaluated by Mr. McKeehan. The results of his study were reported in a paper titled "Fast Reactivity Excursion Device Status Report and Preliminary Design", dated November, 1965.

Method 1

Advantages:
- Outside of core
- Serviceable
- Position indication can be taken from actuator shaft
- Simplicity
- Uses available gas supply
- Reproducible

Disadvantages:
- High reaction load necessary to obtain desirable expulsion times
- Hold-down method necessary to prevent inadvertent expulsion

Method 2

Advantages:
- Outside of core
- Serviceable
- Position indication can be taken from actuator shaft
- Simplicity
- No special hold-down device needed since the pressure is not available under piston until charge is ignited
- Easy to recharge

Disadvantages:
- High reaction loads
- Undesirable gases
- Not reproducible

Method 3

Advantages:
- Safeguards - requires positive input to activate repulsion coil

Disadvantages:
- Complicated
- High impact loads would result in inelastic collision
- A large amount of energy to release in core
- Repulsion coil would have to be very large to obtain the forces needed to accelerate the first mass

Method 4

Advantages: Lighter weight

Disadvantages: In core
Instrumentation would be more difficult
High pressure chanber located in core

After this careful analysis, the method selected for the fast reactivity excursion device was Method 1. A complete analysis of Method 1 is given in Appendix C. The reasons for selecting this method as stated in the G. E. report by Mr. McKeehan in November, 1964, were "simplicity, no new technology needed to develop, serviceable, positive position indication, external to core, and reproducible."

3.3 Design Philosophy

Before describing the details of the design of the FRED as it was envisioned in 1965, a short discussion of "design philosophy" is in order. Since Mr. McKeehan had virtually sole responsibility for the design of FRED, it was his approach to design which probably had the most affect. Very simply stated, the basis of his philosophy was generating questions. This is done by exposing your ideas to different people with varying backgrounds. This was accomplished formally at the Technical Policy Meetings and Safety Audit reports, both discussed previously. The safety audits were conducted by Mr. F. E. Crever, manager of the Technical & Business Planning Operation of the Atomic Products Division. The audits were requested by Dr. K. P. Cohen, General Manager of the Advance Products Operation. The safety audit was primarily in-house. Another less formal in-house procedure was a program of weekly seminars for reviewing designs at the Nuclear Energy Division. Each week the seminar was devoted to a particular project. Mr. McKeehan averaged about one seminar a year. It should be noted that the seminars were more specific in nature while the Technical Policy Meetings were more general. The seminars were attended by anybody who was interested in the project being presented. Mr. McKeehan further noted the most common source of questions is probably the day to day contact between individuals where ideas are exchanged.

There are two points that Mr. McKeehan makes about this approach to design. First, there are some questions that may never be asked that should be. People can ask questions only on information that is given them. Second, one can usually tell how sound his ideas are by the nature and frequency of the questions. If the questions are serious and are brought up often, the designer should be able to tell he has problems with his design.

The designer must continually evaluate the questions being asked. Because of his philosophy, Mr. McKeehan did not view these meetings and reports as an imposition but as a chance for communication.

Another point that Mr. McKeehan made was that some designers like to do detailed analysis while others prefer to take a design all the way through. He does not require his people to do their own analysis, for there are people within the division who enjoy and are equipped for this type of work. There are also special groups set up to handle particular problems such as stress analysis.

3.4 Maximum Hypothetical Accident

An important aspect of the design of the FRED device was the Maximum Hypothetical Accident (MHA) or Designed Basis Accident (DBA). Currently at G. E. it is called the Hypothetical Core Disassembly Accident (HCDA). A good definition of an MHA is quoted below:[10]

> The Maximum Hypothetical Accident refers to the set of abnormal conditions which lead to calculated energy release and release of radioactive products believed to be an upper limit physically possible in the SEFOR system. The MHA is used as a test of the containment both from the standpoint of kinetic effects and as the initiating event leading to potential release of radioactive materials. Many of the assumptions used to calculate the MHA are extreme and represent bounding cases.

Two types of limiting conditions which could result in an MHA were hypothesized. Quoting from the source above, these were: "(1) A sudden increase in core power density followed by loss of coolant leading to a rapid melting of fuel in the core and compaction of the core into a super-critical mass [Core Meltdown Accident], and (2) A sudden increase in core power density followed by the rapid insertion of reactivity [removal of the poison slug] in excess of $1 by the FRED when the core is operating well above the maximum design temperature limit." The events leading up to the first and second MHA's are listed in Figures 3.5 and 3.6 respectively. The term "scram" used in the figures refers to the removal of the reflector from around the core. The reflector normally reflects neutrons back into the core. When scram occurs, neutrons are

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[10] Facility Description and Safety Analysis Report, Volume II, Section 16.4.1.1.

CORE MELTDOWN DISASSEMBLY

SEQUENCE OF EVENTS & ASSUMPTIONS

1. THE FRED IS FIRED
2. THE CORE COOLANT FLOW STOPS
3. NO IMMEDIATE REACTOR SCRAM
4. FUEL MELTS
5. FUEL CLADDING MELTS
6. FUEL COLLAPSES UNDER GRAVITY
7. REACTOR SCRAMS

FIGURE 3.5 - Core Meltdown Disassembly

DISASSEMBLY DUE TO "FRED"

SEQUENCE OF EVENTS & ASSUMPTIONS

1. THE CORE IS OVERLOADED
2. THE FRED IS CHARGED
3. CONTROL ROD INTERLOCKS FAIL
4. REFLECTOR RUN-IN @ 10¢/Sec.
5. NO REACTOR SCRAM
6. FUEL HEATS UP TO 9000°F
7. THE FRED IS FIRED

FIGURE 3.6 - Disassembly due to FRED

allowed to escape from the core and a chain reaction can no longer be maintained. In the SEFOR reactor, the reflector drops below the core during scram. If scram occurred after the fuel melted, the reflector would again be surrounding the fuel.

3.5 Rate of Reactivity

The core meltdown accident was chosen as the MHA. The next step was to determine the rate of reactivity insertion caused by core meltdown and design the FRED so it would not exceed this value. The change in reactivity was 3$/inch change of fuel height as it slumps (melts) and fills the void normally occupied by the sodium coolant. The fuel meltdown would begin in the center of the core and would expand radially. Therefore, for purpose of analysis, the core wss divided into annular rings containing integral numbers of fuel rods. The reactivity insertion rate was determined by combining the reactivity worths of an individual ring and the time that the ring would begin to melt. Time would be based on heat transfer calculations. The results are plotted in Fig. 3.7. The maximum rate is 50$/sec and would occur 0.2 sec after start of core meltdown. This reactivity insertion rate was used in the design analysis of the SEFOR and found to be safe.

The General Manager then set the FRED insertion rate at 20$/sec as the design limit. This was a management decision. If this limit was inadequate for test purposes, it was up to the Advanced Reactor Physics group to justify an increase in rate. As it turned out, those concerned were able to agree on this rate since they lost only 5-10% in accuracy in experiment when they were looking for plus or minus 25%. The 50$/sec was the highest rate they did the analysis for, because they did not have enough funding to study the effects of higher rates. They were criticized for this at the AEC licensing meeting because they did not know the threshold of damage rate.

3.6 Observations Concerning the MHA

Mr. McKeehan made some interesting observations about the implications of an MHA when asked his opinion of it. He felt that the hypothetical accident is strictly hypothetical. To arrive at an MHA requires a host of unusual events taking place in the proper sequence. It is not simply a seal failing. He feels that in some ways it has been bad for the nuclear industry to hypothesize these accidents. Opponents of nuclear power tend to look at the MHA as an industrial admission that if all these things occur the reactor is going to "blow up" and release large amounts of radiation to the environment.

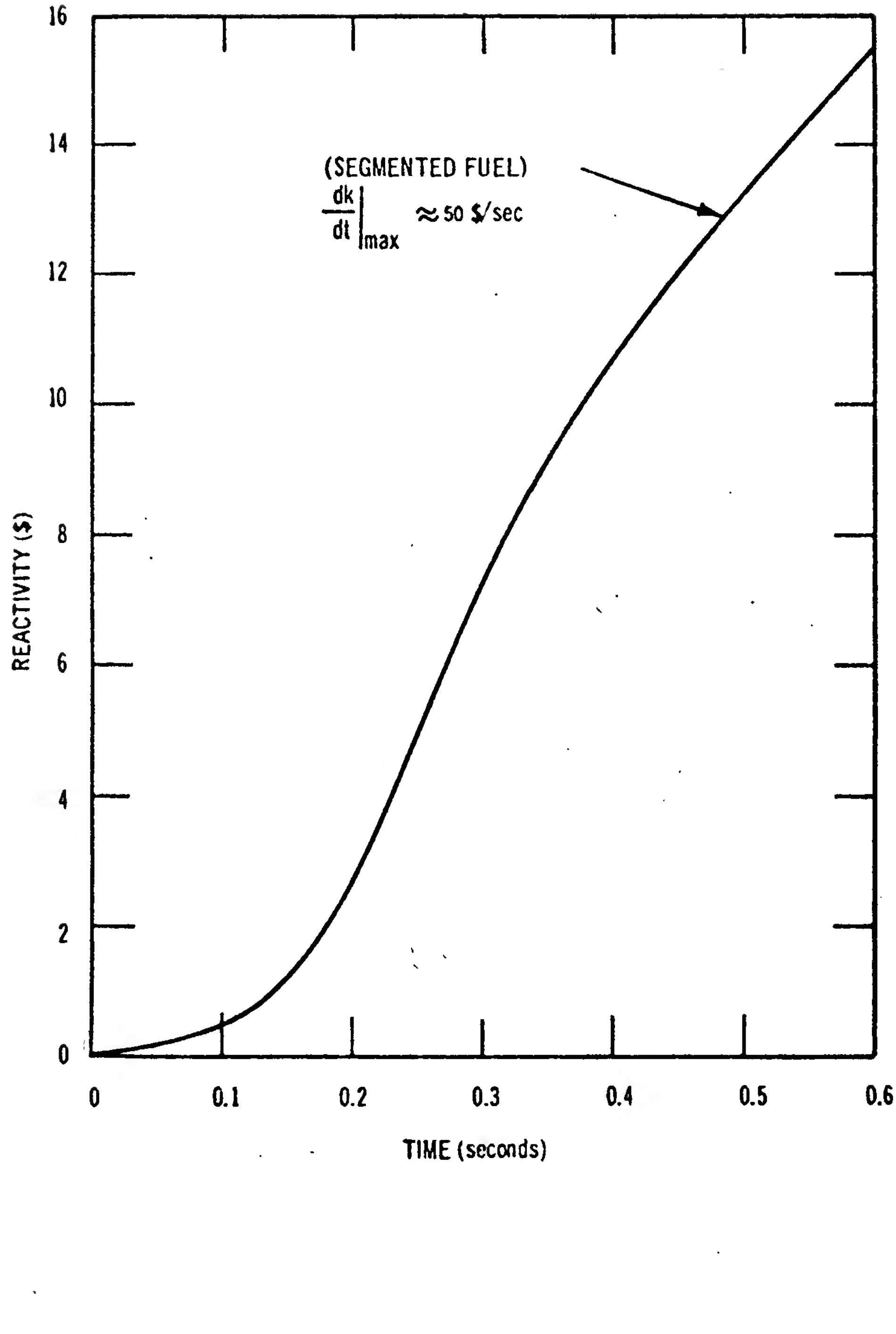

FIGURE 3.7 - REACTIVITY INSERTION BY CORE MELTDOWN

The nuclear industry is not saying that the reactor is going to "blow up". The occurrence of the proper sequence of events leading to the MHA is extremely remote. It is Mr. McKeehan's feeling that theorizing the MHA is necessary as part of a disciplined approach to engineering a reactor plant. He further states that designing for an MHA is like designing a car in which a person won't be killed in an accident, when he is drunk, driving on the wrong side of the freeway and it's snowing in California. It is the same idea although the magnitude of an accident is much more serious in the nuclear industry. What he feels is happening is that by concentrating on the hypothetical accident, the designer might actually compromise the design. That is, he might lose sight of the day-to-day incident which is more likely to occur. Again he uses the automobile to illustrate what he means. The designer might design the car so the driver will survive a 100 mph collision with a wall, but doesn't include brakes because he forgets about the day to day requirement to stop at a stop sign. Mr. McKeehan stated that the outer containment structure must be designed to contain any MHA, but most other equipment should be designed for some lower level accident, one that is more realistic and likely to occur.

4. FRED MODEL 1

4.1 Description

The FRED Model 1 is shown in Fig. 4.1. The following is a description of the FRED Model 1 and how it was to work.[11] The upper chamber above the piston (guide tube) was charged with a low pressure gas; this would hold down the piston as the lower chamber (expansion chamber) was charged with the high pressure gas. This is achieved since the area of the piston exposed to the high pressure is only that of the nozzle orifice while the whole piston top area is exposed to the low pressure. The areas involved were such that they had a ratio of 20 to 1. At a preset pressure in the high pressure tank, the exhaust valve would open and vent the pressure in the guide tube to atmosphere. As already described in method 1, the piston would then be exposed to a sudden force. The piston was attached to an actuator rod which was attached to the poison slug. The poison slug was positioned in the center of the core when the piston was in the down position. One of the reasons for charging the chamber above the piston with low pressure gas was the initial experimental SEFOR plan calling for a range of piston speeds from 0.025 sec to 1 sec. It was felt that they could vary this pressure to achieve the desired travel time range.

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[11] Information in this section taken from SEFOR Development Program Third Quarterly Report, DEAP-4799, October 1960-January 1965.

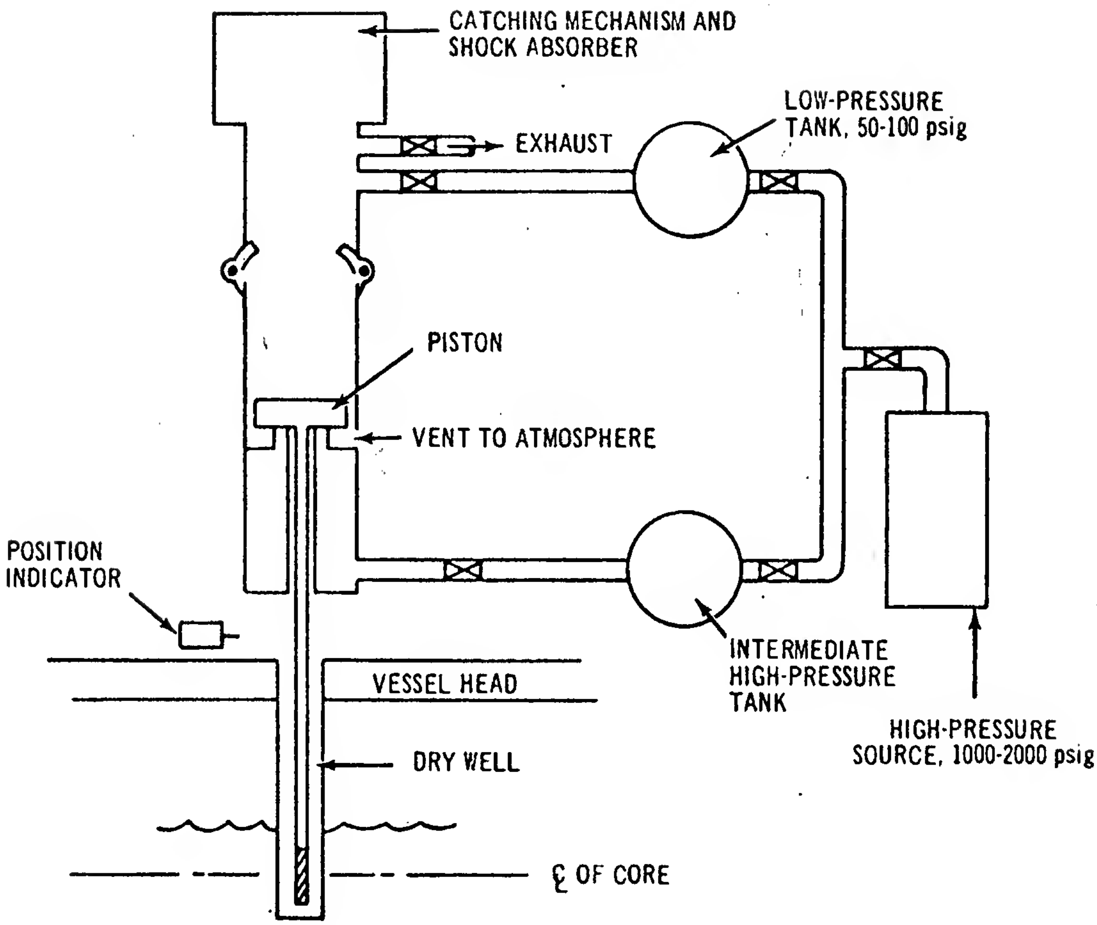

FIGURE 4.1 - FRED Model 1

The latches at the top of the guide tube were to prevent the piston from falling back after the FRED was fired. Mr. McKeehan felt at this time that if the piston were allowed to fall back it might damage the nozzle orifice seal. When the piston reached the top of the guide tube, the piston would be stopped by a hydraulic dashpot.

It was desired that the position of the piston be known during an experiment for data reduction purposes. To accomplish this a photoelectric system was used. The photo cell detected the changes between the dark and light stripes painted on the actuator rod (rod between piston and poison slug). These stripes were about one inch thick except for three dark stripes that were three inches long to provide a data reference point. The cell is located just below the expansion chamber and is shown in Fig. 4.2.

The FRED was to be located on top of the reactor core, and the poison slug was to be positioned at the center of the core from which position it was to be fired during the SEFOR experiments. Total travel was to be about 26 inches. An additional mechanism that the FRED was mounted on was a positioner mechanism. This simple screw jack mechanism was to be used with the oscillator in the oscillation tests[12] to be conducted before the FRED was to be used. The positioner would make it possible to position the FRED and poison slug above the core so that preliminary on-site tests could be conducted without affecting core reactivity. The poison slug was to be inserted into a drywell, thus eliminating variations caused by viscous drag since the core was to be cooled by liquid sodium.

The high pressure source was to come from an accumulator with a liquid volume of 70 in^3. Because of its small size, it was excluded from the ASME Pressure Vessel Code but it was designed so that it would meet code stress levels. The expansion chamber was designed for 1100 psi but only 200 psi was expected. It had a liquid volume of 50 in^3. The guide tube was attached to the expansion chamber. The inside of the tube was nitrided, though no significant wear was expected.

Most of the parts for the FRED were manufactured from type 304 stainless steel. The exceptions were the piston and orifice piece. The piston was made from low carbon steel and the orifice piece from type 410 stainless steel.

The gas used in the system was Argon. Argon was selected because it is a noble gas.

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[12]These were quasi-static tests in which the core reactivity was sinusoidally oscillated, and was another method used to measure the Doppler coefficient.

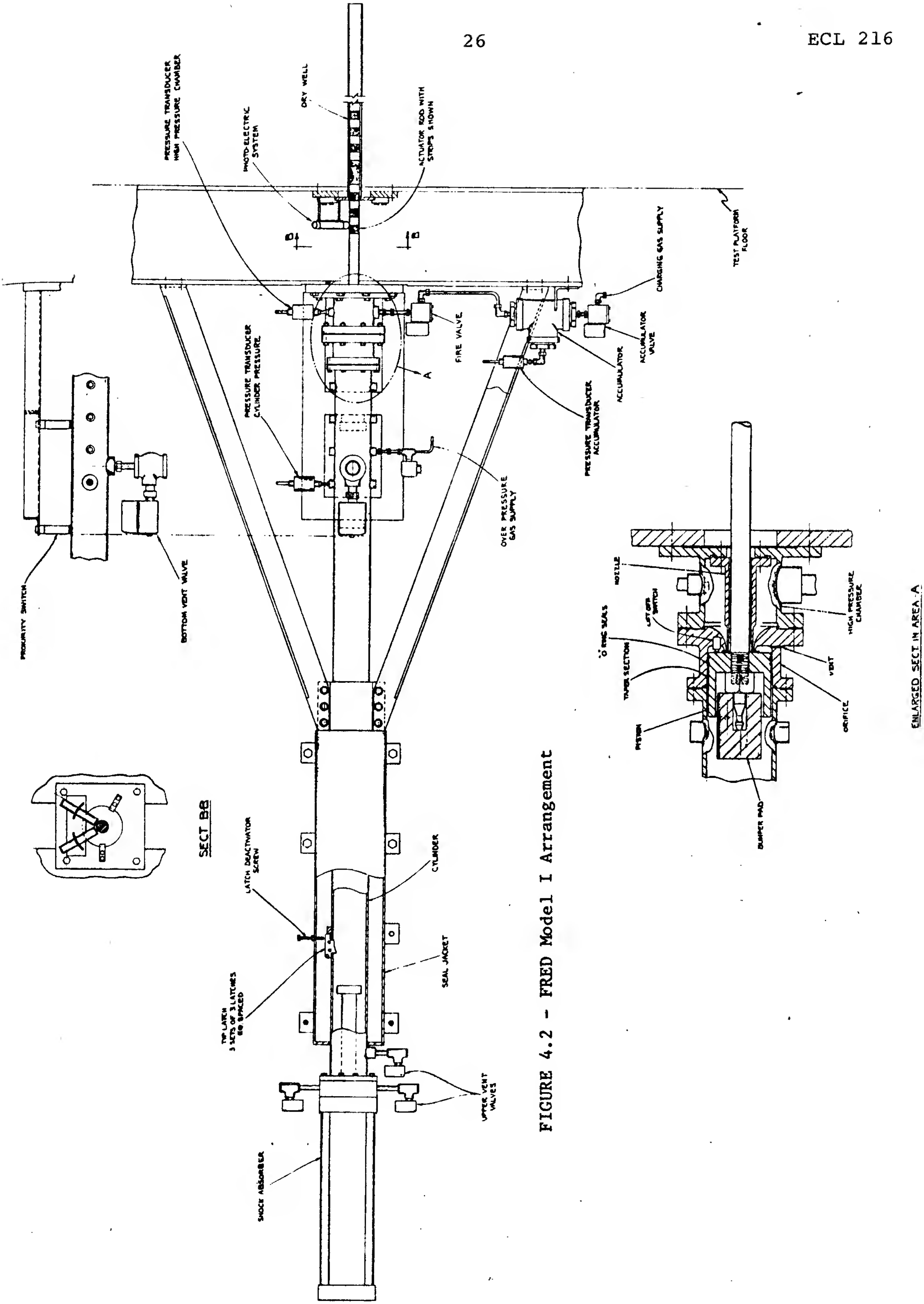

FIGURE 4.2 - FRED Model I Arrangement

4.2 Fabrication

During the year after the initiation of the SEFOR program, most of the FRED was designed and fabricated. With the exception of nuts, bolts, screws, hose, solenoid, and photo cells purchased from vendors, the system was designed and built by G. E.

Most of the fabrication of the FRED was done in the G. E. shops at San Jose. The only exception was the upper guide tube section which was made by a local machine shop. Mr. McKeehan did the necessary liaison work. The piece was assembled at the G. E. shop.

The assembly of the FRED was carried out not by shop personnel but by technicians working for Mr. McKeehan. The reason for this was that the technicians were the ones who were going to be doing the necessary design changes and modifications during testing. It was necessary that they become familiar with the mechanism. Having technicians assemble the FRED had its advantages. Mr. McKeehan pointed out that the technician is usually a highly skilled and motivated individual who has a certain amount of interest in his work and the design. He will take his time and do a good job when assembling a mechanism. The average individual in the shop tends to be less careful and parts may get banged around and damaged. What can happen to a designer is that his design may be a good one in the hands of a technician but appear to be faulty when the shop personnel assemble it.

4.3 Testing Model 1

Testing of the FRED Model 1 took about a year starting in early 1966 and was done at G. E. in San Jose. The program was planned to test the dynamics of the FRED and the results would be compared to that of the analytical model. Another purpose for testing was to change various paramaters and then determine the effects this had on travel time. The parameters which were varied were flow areas, weight of moving parts, venting in the guide tube, expansion chamber volume, and pressure and type of gas.

The test procedure was as follows[13]:

1. Pressurize the guide cylinder over the piston through the over pressure gas supply line.
2. Pressurize the expansion chamber by opening both the fire valve and accumulator valve with the pressure control provided by a preset pressure regulator.
3. Close fire valve

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[13] Design and Testing of the SEFOR Fast Reactivity Excursion Device, GEAP-13649, October 1970.

4. Continue to pressurize the accumulator to the desired pressure. (It is always higher than the pressure in the expansion chamber).
5. Close accumulator valve.
6. Start data-recording system.
7. Turn fire switch.
8. Stop data-recording system.

Figures 4.3, 4.4, and 4.5 show the results of varying the expansion chamber pressure, the weight and connecting hose diameter, and the type of gas, respectively. Fig. 4.6 shows that effect of changing the number of vents in the guide tube from none to two. The same figure shows the effect of changing the volume of the expansion chamber from 50 in^3 to 25 in^3. This was done by filling the chamber with aluminum filler rings. The smaller chamber produced slightly faster exit time. But as can be seen, none of the results are changed significantly.

From the analytical model, it was predicted the FRED would be self-limiting. That is, the change in exit time would be small due to pressure variations in the accumulator in the region where FRED is operated. Fig. 4.7 shows the experimental and the analytical results, confirming the self-limiting characteristics of FRED.

4.4 Initial Design Changes

A number of minor changes were made on the FRED during this testing program, as the tests showed some original ideas could be improved. One thing the tests showed was that a reasonable range of exit times could be obtained by utilizing only the accumulator as a means of firing. The system was made much simpler by eliminating the pressurization of the guide tube which had not worked as expected. It took more pressure to hold down the piston than calculated by the ratio of areas. The pressure ratio turned out to be 8 to 1 due to probable leakage around the nozzle orifice. Secondly, the hydraulic dashpot was eliminated because it was found that the piston was slowed down by compression of the air trapped above it.

The fact that the piston was slowed sufficiently by gas compression lead to removing the latches at the top of the guide tube. This allowed the poison slug to drop back into the core and act as an additional safety measure. If an accident should occur, the poison would be reintroduced into the core to slow the chain reaction. The gas on the underside of the piston would be slowly vented between the actuator rod and expansion chamber. The poison slug would take 1 to 5 seconds to drop back into the core, well after the transient effects were completed and the reactor scram took place.

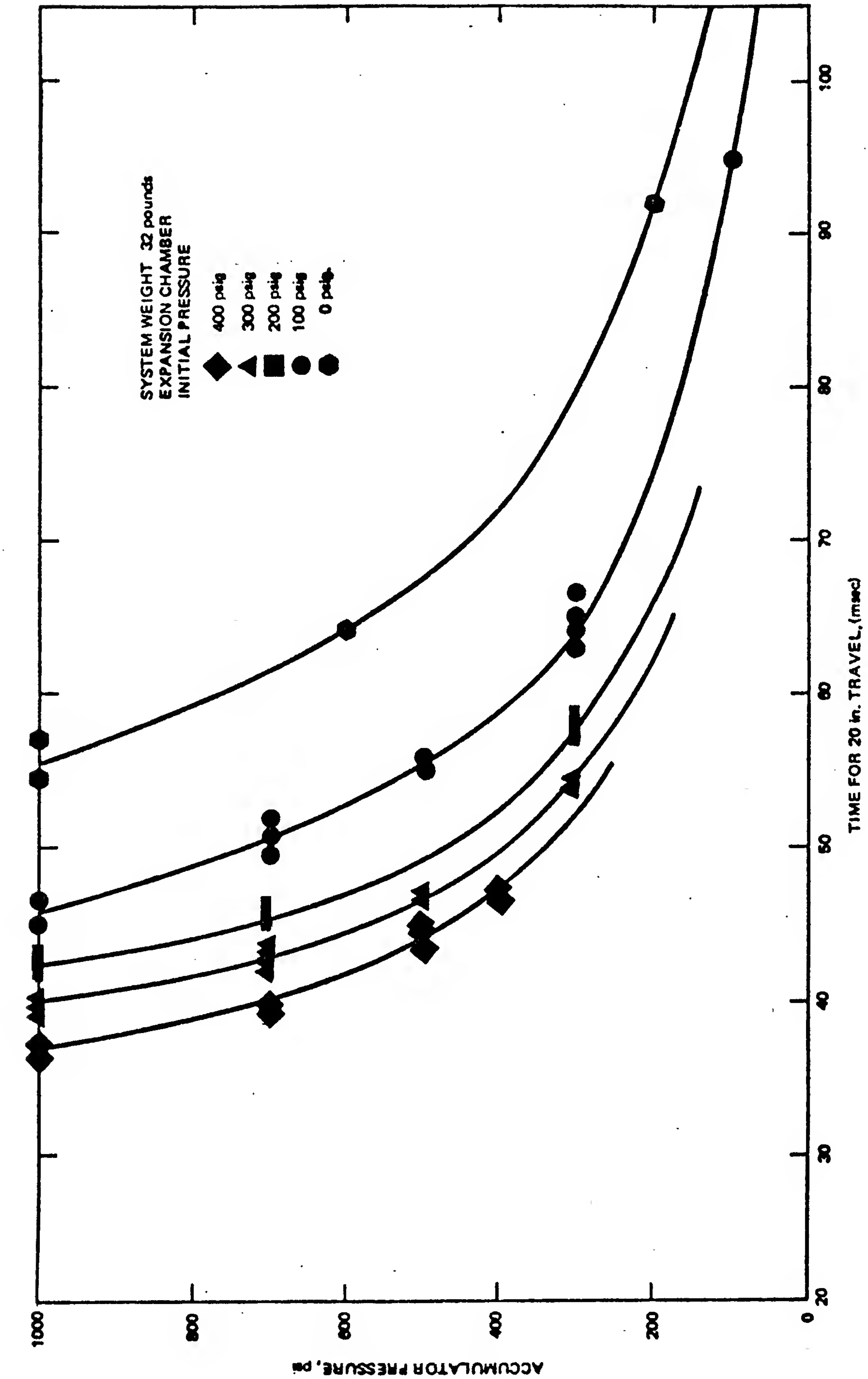


FIGURE 4.3 - FRED Dynamics

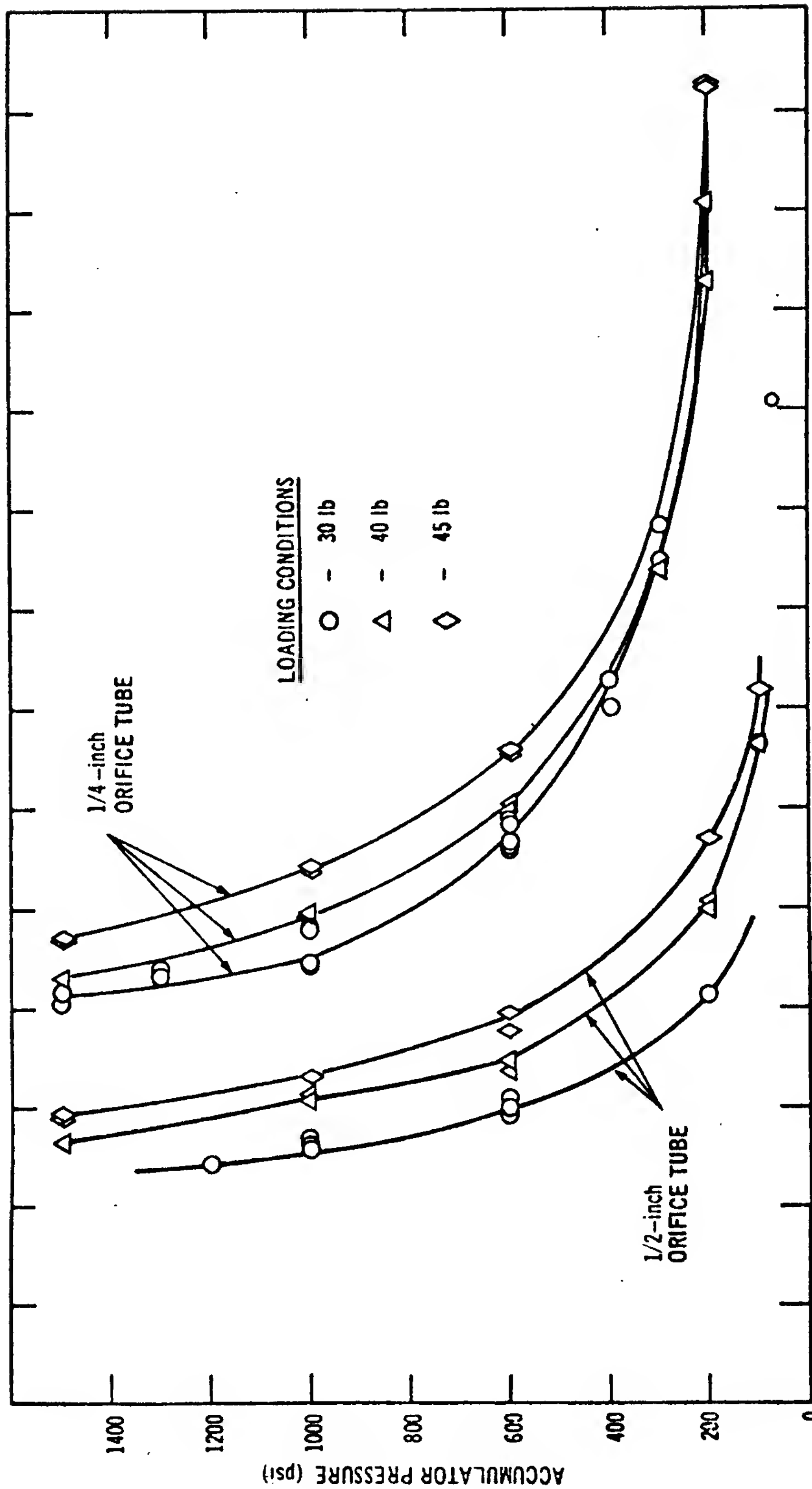


FIGURE 4.4 - FRED Testing

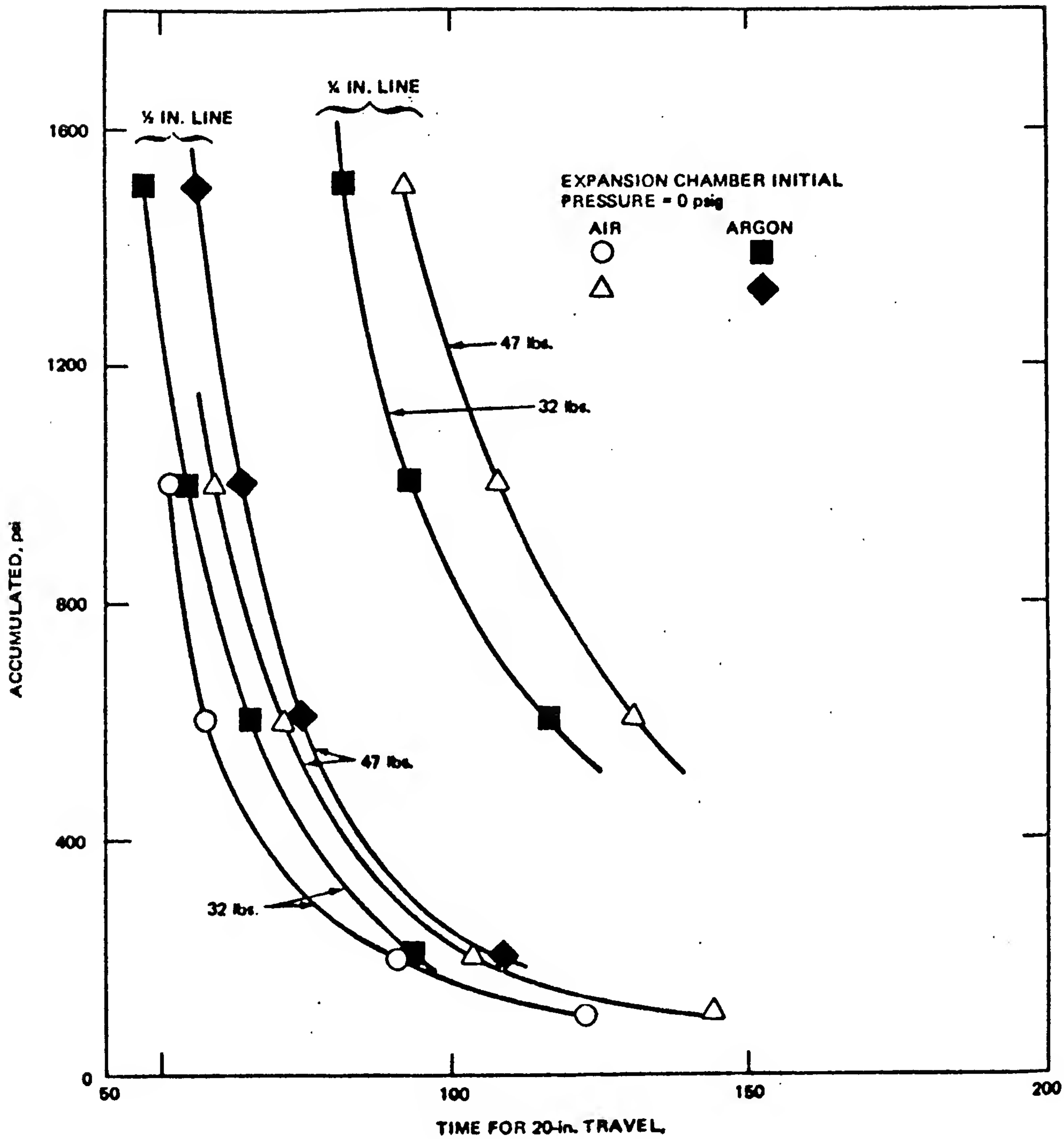


FIGURE 4.5 - Effect of Changing Gas

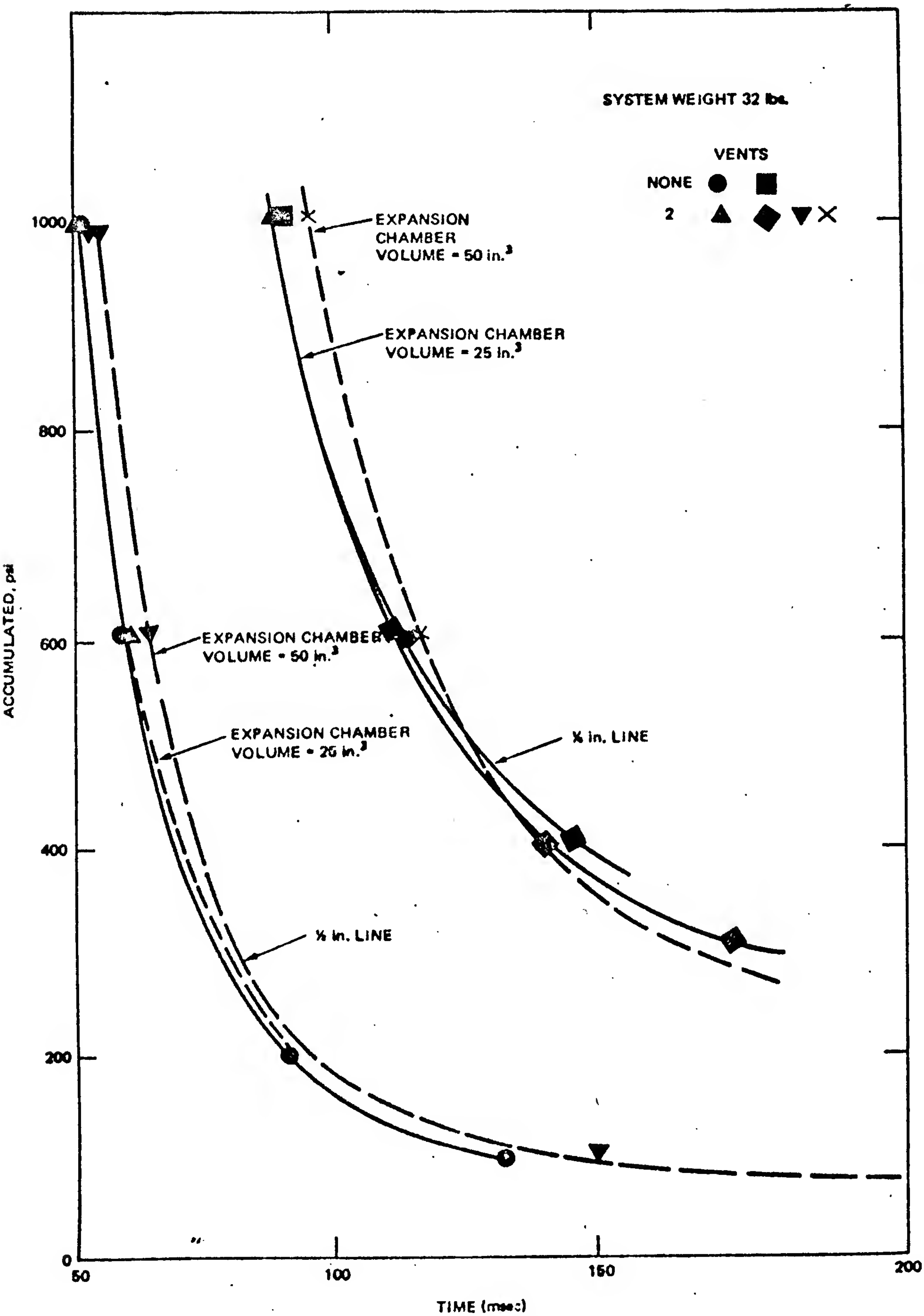


FIGURE 4.6 - Effect of Changing Expansion Chamber Volume

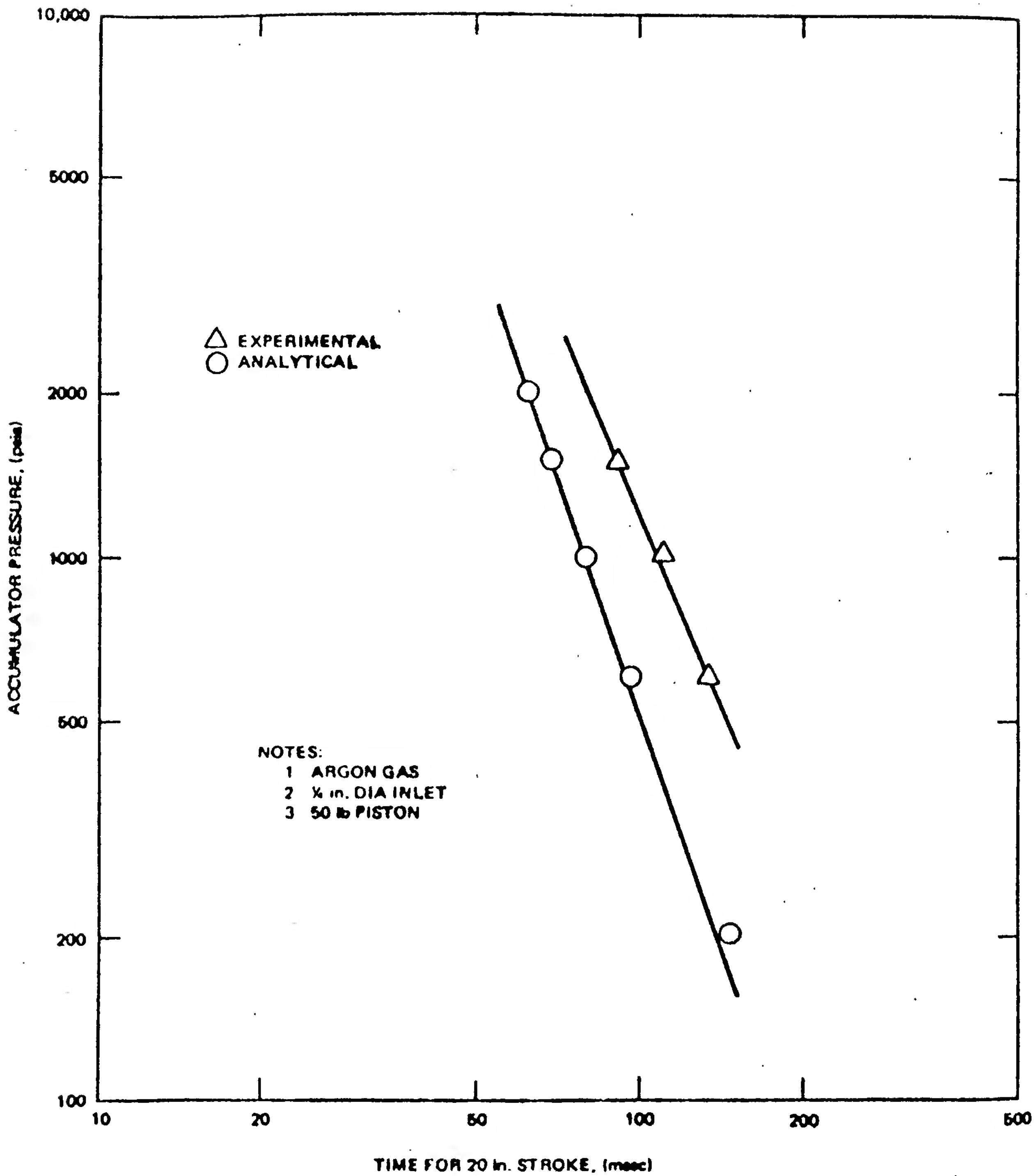


FIGURE 4.7 - FRED Characteristics

4.5 Change of Functional Requirements

Although some important modifications were made on the FRED during the testing of Model 1, most of the original design proved to be satisfactory. Mr. McKeehan explained that he was cautious enough in his goals when he started the project to be sure that he could meet them. Although they had not achieved exit times of 25 m sec, they were well on their way to meeting that goal when the functional requirements were changed.

The eventual exit time requirement that was designed for was 90 m sec and a much smaller range (up to 150 m sec). There were a number of reasons for this change in requirements. The most important was simply that the old requirements were no longer valid. The change was brought about when the project manager, Dr. Wolf, questioned the nuclear engineering people on their requirements. They were the "experimental specifiers". Mr. McKeehan had the job of incorporating the physical requirements into the design so that the desired results would be obtained. The nuclear engineering people were able to relax their requirements as their analytical model evolved and they could more accurately predict SEFOR response.

Since SEFOR was a development program, the goals were set intentionally high and were reduced only when they proved unworkable. The procedure for changing these requirements was both formal and informal. Formal documentation was required when such things as safety, results, or expense would be significantly affected by a change in requirements. As it turned out, increasing exit time was not a significant change; had it been necessary to reduce the exit time a formal procedure would have been required. By the time the SEFOR experimental program was started, the analytical model was good enough so that there was very little difference between the analytical and the experimental models.

4.6 Safety Considerations

A number of questions raised about safety concerned either an over-pressurization of the FRED or inadvertent firing of the FRED. The first situation would not be too serious a problem because of the self-limiting nature of FRED. The minimum exit time that the FRED could be fired when over-pressurized from the system was 80 m sec using a 1/4" charging line. To alleviate any questions about possible rupture disk failures, two more disks were added to the accumulator chamber.

An interesting comment was made by Mr. McKeehan about the people with reliability. One might think that the goals of both groups are always the same, but this is not always the case. When the safety group might want to add detection or safety devices, the reliability group becomes concerned about the reduced reliability because of the additional equipment.

To eliminate the inadvertent firing of FRED, three safety latches were installed in the guide tube just above the piston as shown in Fig. 4.8. Any one latch could hold the piston if the FRED was inadvertently fired, but some distortion would result. The latches were designed so that if the piston were forced against them they could not be withdrawn, as shown in Fig. 4.8. The latches were independently controlled and monitored. Limit switches were used to indicate if the latches were in the tube or not. As a further safety measure, the power to the latches is insufficient to withdraw them simultaneously.

4.7 Final Design Changes

The immediate results of the test program for the FRED 53 Model 1 as mentioned previously, were the elimination of pressurizing the guide tube, the decision to remove the latches at the top of the guide tube, and the elimination of the hydraulic dashpot. To prevent possible damage to the orifice nozzle when the piston fell back, a rubber seal cushion was added to the bottom of the piston. To slow the fall of the piston, an additional labyrinth seal was added between the actuator shaft and the expansion chamber as shown in Fig. 4.9. To achieve a fast acting system, a 28 Vdc, solenoid actuated balance poppet three-way valve was used. When de-energized, it would vent the expansion chamber to the cell atmosphere. When energized, it would fire the FRED. To replace the dashpot, a cushion chamber was designed. This was simply an extension of the guide tube with a series of holes drilled into the side to allow venting as the piston moved up. The holes were made 1/16 inch, with the idea that they could be enlarged if venting was insufficient and the piston was slowed too much. As it turned out, the holes were adequate and did not have to be enlarged.

A further refinement to the position indication system was made by the addition of two lift-off switches and a variable inductance sensor. These were located at the bottom of the nozzle orifice below the piston as seen in Fig. 4.9. The switches would provide the primary signal for start of motion, while the variable inductance sensor would provide analog data of slight piston movement before piston lift-off. This information was strictly for data reduction purposes.

These design modifications were incorporated into a new FRED model which was to be the reactor model used in SEFOR. The reactor model is shown in Figures 4.10, 4.11, and 4.12.

4.8 Poison Slug Worth

The poison slug worth was another important design problem that will now be considered. The poison sections were to be about 15 inches long and were to be filled with boron carbide (B_4C). All poison slugs were to be within 1/2 lb of each other. Their reactivity worth was electro-etched on the outside surface. Nuclear Engineering determined the required reactivity worth of

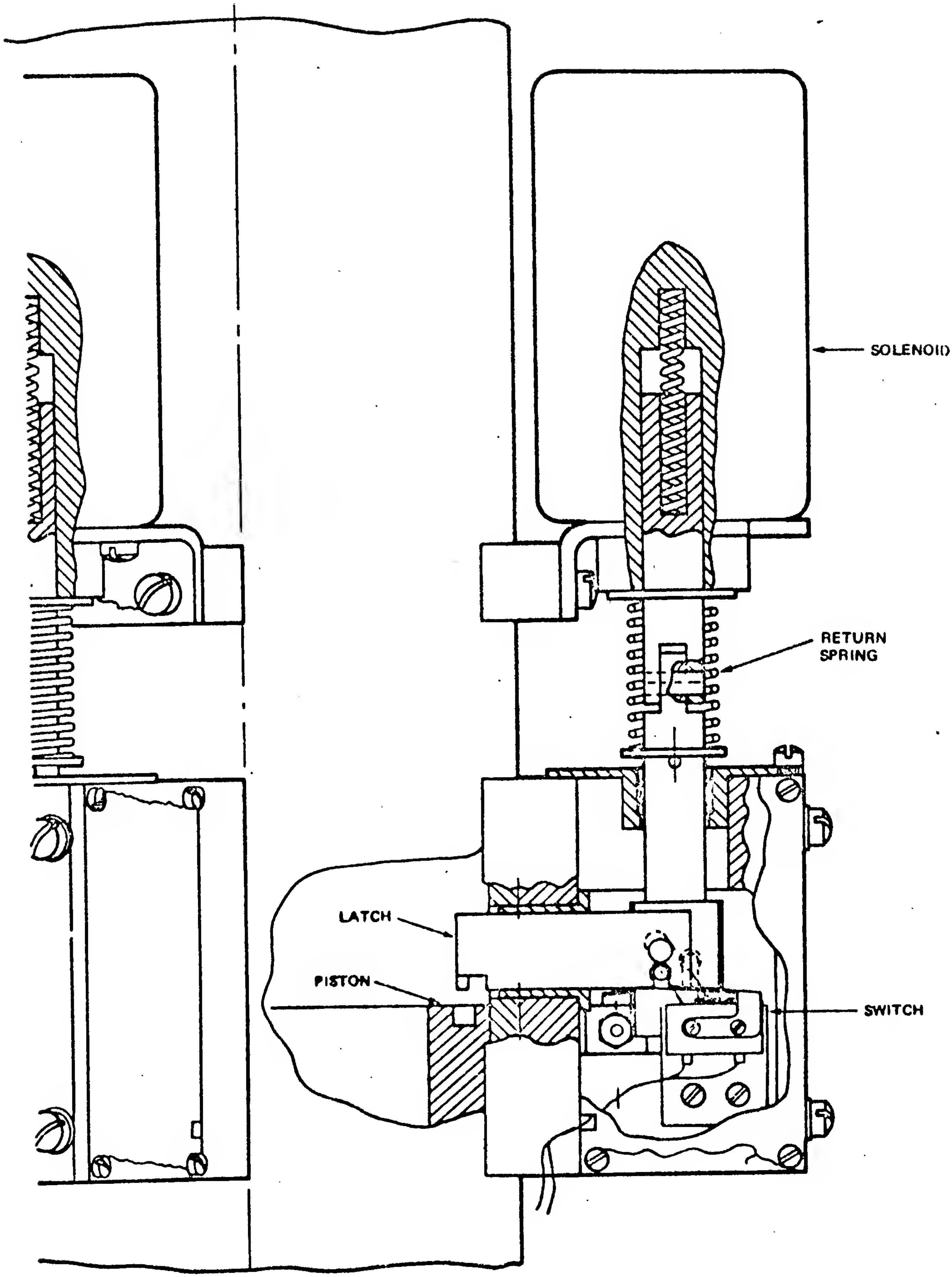


FIGURE 4.8 - Safety Latch Mechanism

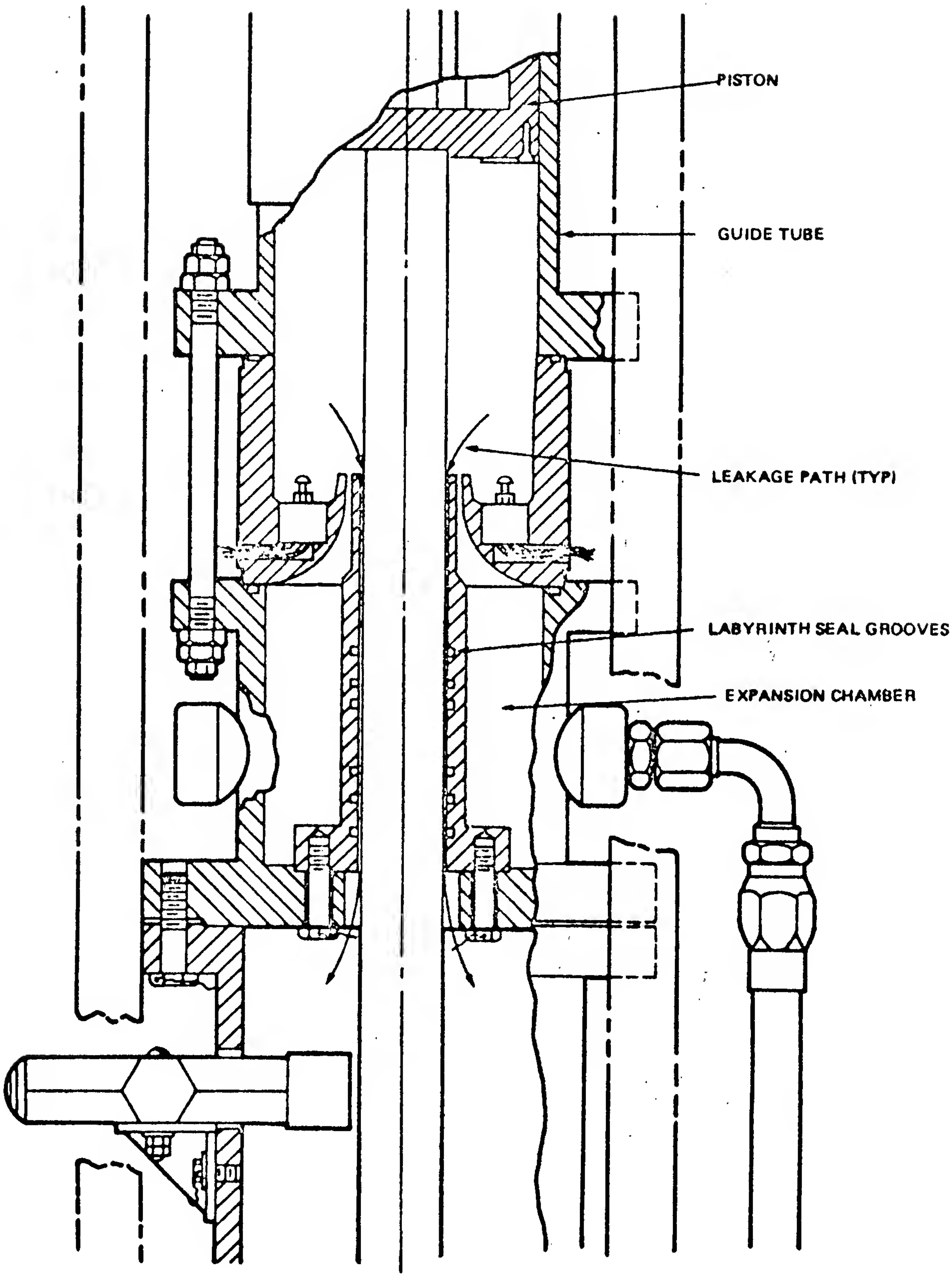


FIGURE 4.9 - Drop Back Cushion

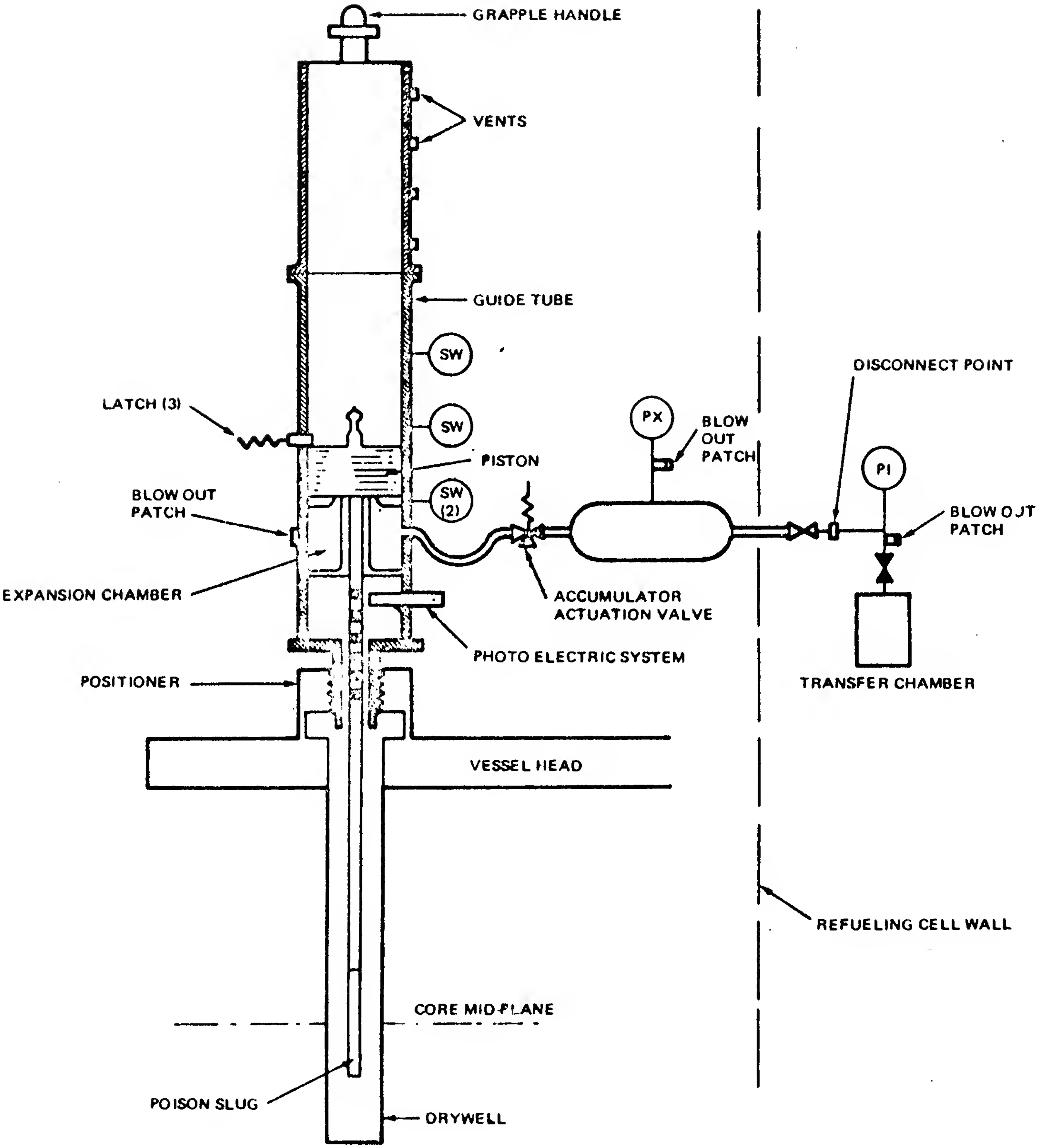


FIGURE 4.10 - FRED Schematic

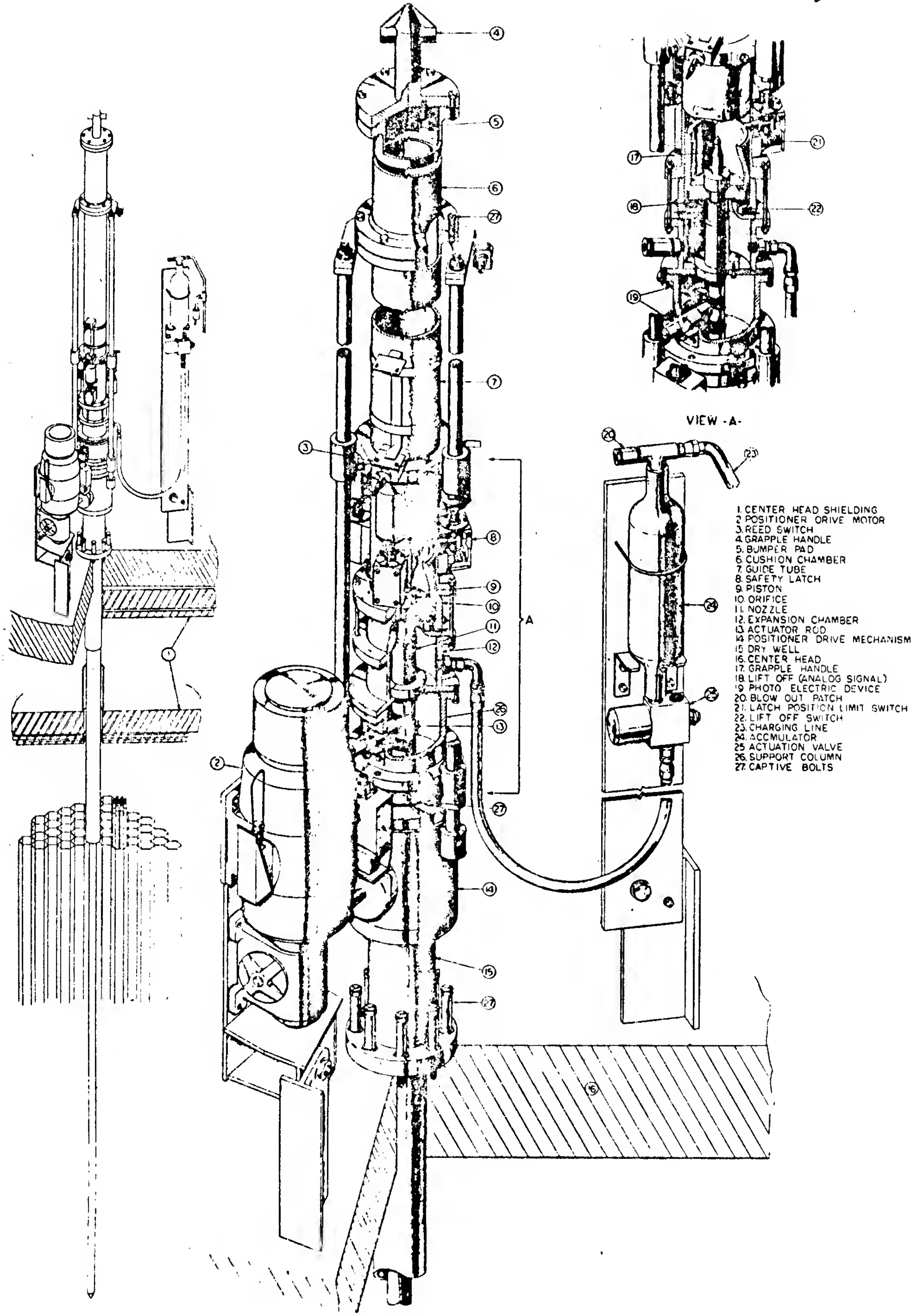


FIGURE 4.11 - Fast Reactivity Excursion Device

FIGURE 4.12 - FRED Reactor Model Test Setup

each poison slug for each SEFOR experiment. They initially asked for a slug worth of 99.5¢ which was beyond what engineering was (economically) capable of producing. They could produce slug worths within ±10¢.

To achieve the required slug worth, the poison slug with its rod assembly was manufactured as close to the desired worth as possible. The poison rod assembly was composed of three subassemblies as shown in Fig. 4.13. They are: the handle, the B_4C assembly, and the bottom nose piece. The rod assembly was shipped to the SEFOR site and placed in the core. The slug worth was then determined at low power. The slug worth could be changed by changing the position of the slug with respect to the core center line. If an adjustment had to be made, the slug assembly would be shipped back to San Jose. The welds at 4 and 5, Fig. 4.13, would be machined out and the handle would be lengthened or shortened while a piece was added or subtracted from the bottom nose piece. The modification of the nose was done to maintain the same length and weight. This gave them an adjustment of 6 inches which could affect the slug worth as much as 20%.

5. REACTOR MODEL TESTS

In October 1968 a series of tests was begun on the Reactor Model. The first tests were conducted to determine the connecting hose size. The hose was located between the accumulator and the expansion chamber. The results are shown in Fig. 5.1. It was decided to use the 1/4 inch diameter hose. The second series of tests was conducted to determine if the piston would contact the bumper pad located at the upper end of the cushion chamber. This was done by installing two proximity switches, one 2 inches below the bumper and the other 12 inches below it. Tests were conducted up to 2000 psig and the piston never reached the 12 inch level. It was also observed that the piston was being effectively cushioned when it fell back to the nozzle orifice. Since previous tests were conducted at room temperature, tests were conducted to determine if the reactor temperature environment would significantly alter the FRED dynamics. No significant differences were observed.

To substantiate the safety characteristics of the FRED, a number of additional tests were conducted. Fig. 5.2 again is an illustration of the self-limiting characteristic of the FRED. To establish FRED characteristics under abnormal conditions, the bottom of the drywell was deflected as much as 3/4 of an inch before firing the FRED. The misalignment showed no significant effects of the FRED dynamics. The FRED was also fired against safety latches with no apparent damage. Tests were also conducted to test the rupture disks.

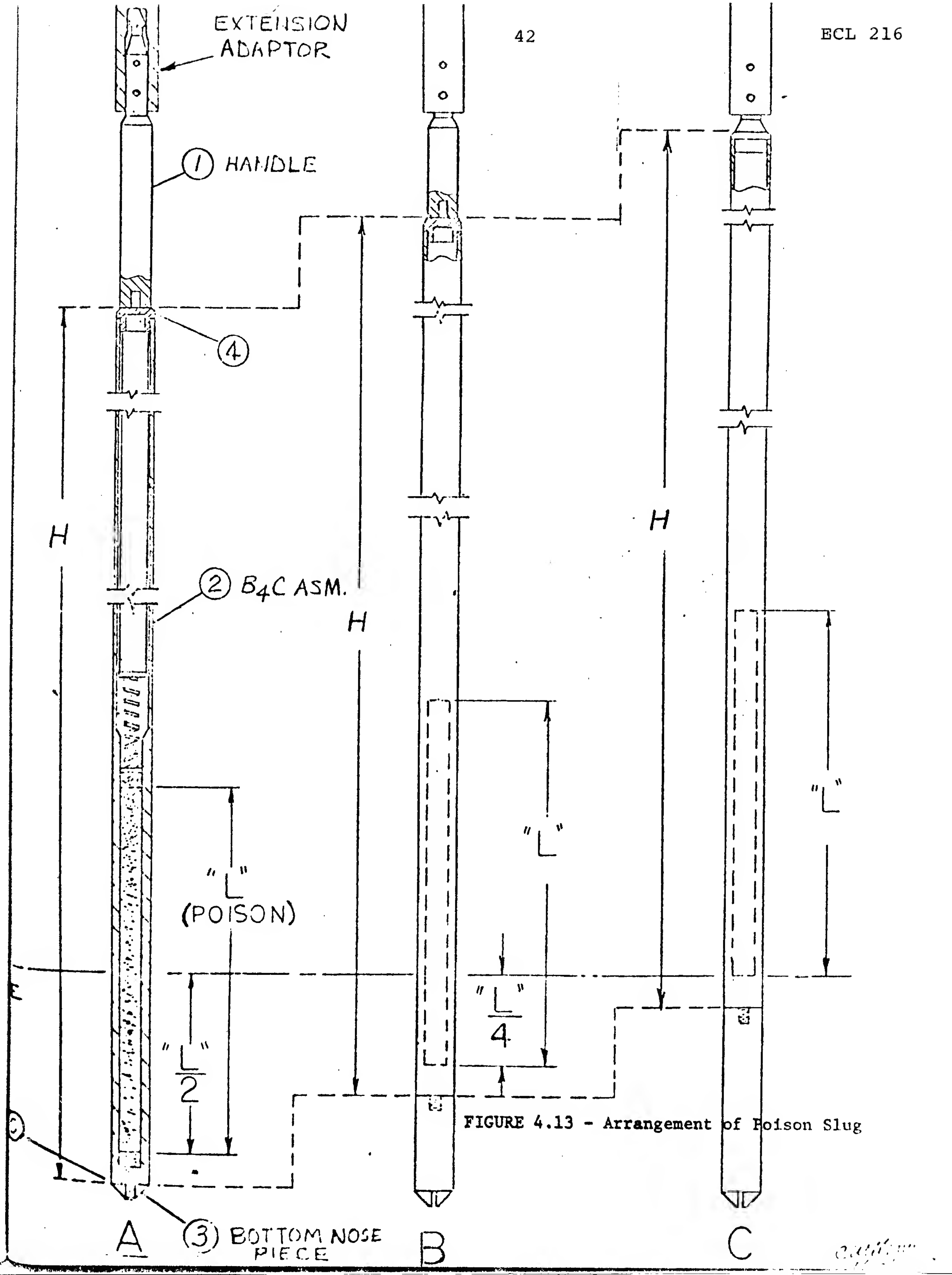


FIGURE 4.13 - Arrangement of Poison Slug

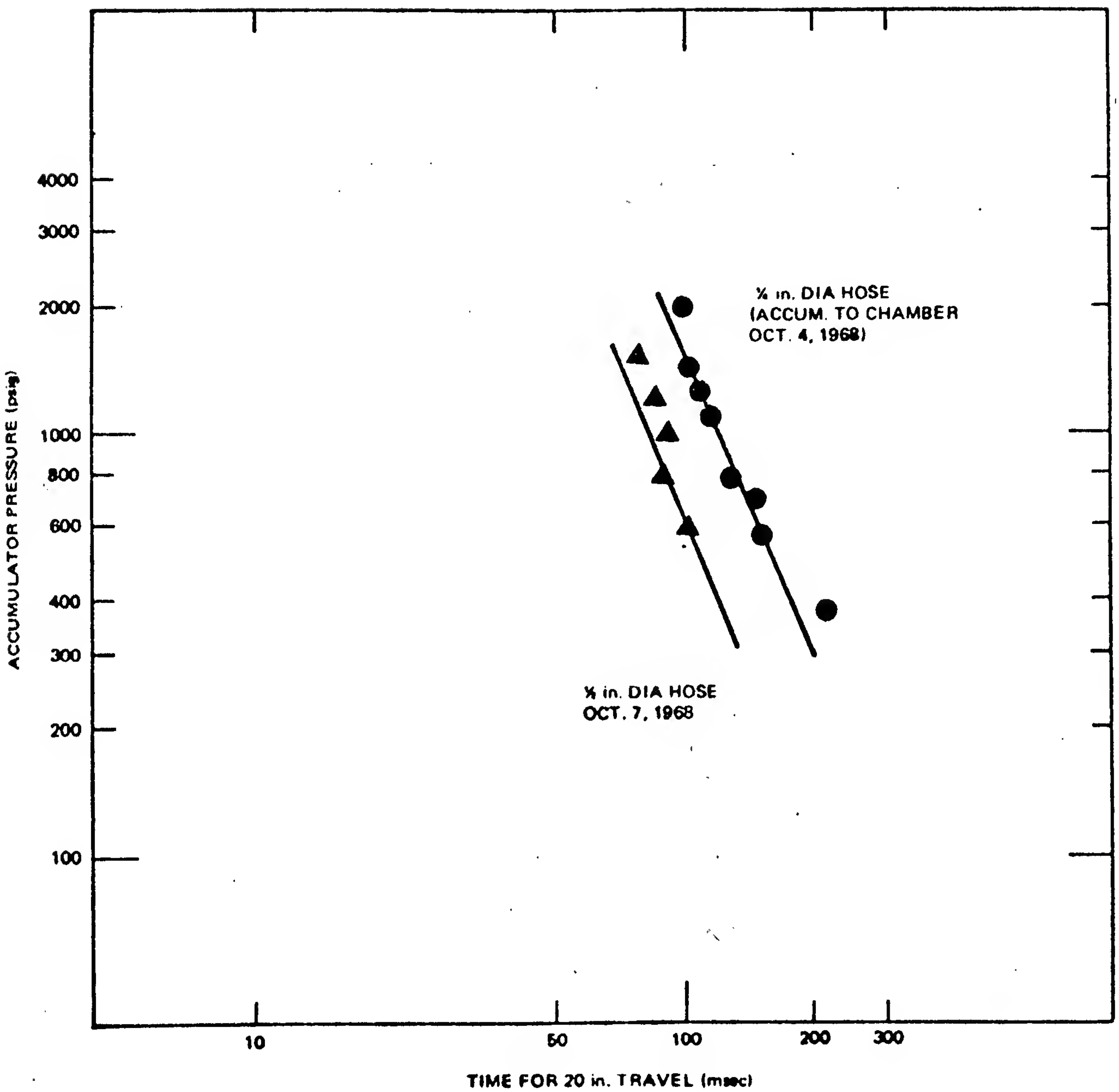


FIGURE 5.1 - Reactor Model Tests with 1/2" and 1/4" Hose

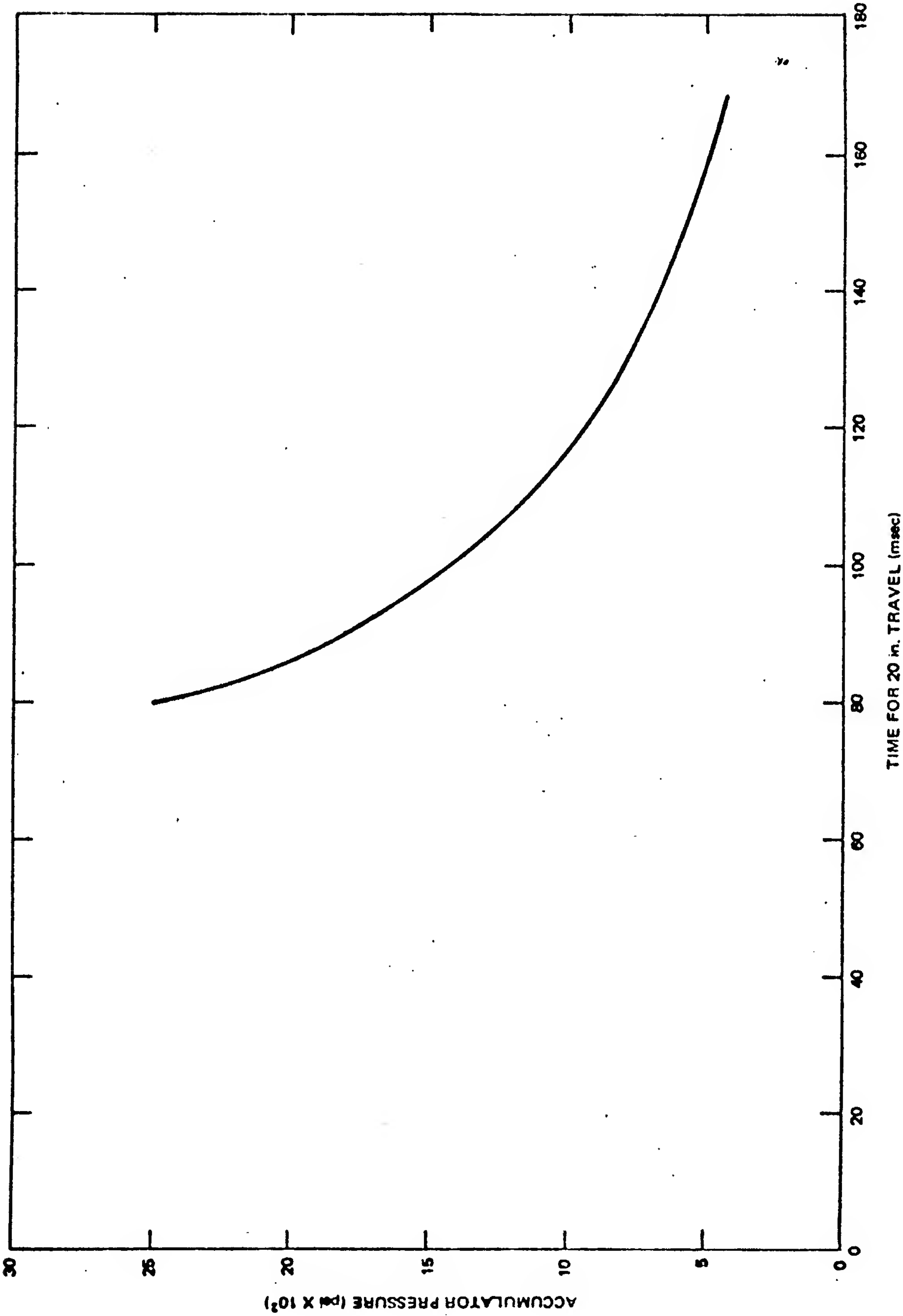


FIGURE 5.2 - FRED Self-Limiting Characteristics

One test which has not previously been mentioned involved holding the FRED down when it was fired. When the pressure had built up, the piston was released, thus getting a much faster exit time. They planned to hold the piston down by rope and when the pressure had built up, they hoped to cut the rope, but so much force was involved that the rope would break before they could cut it.

This was an example of a question which should have been asked, but was missed until near the end of the project. The question was: Wouldn't much faster exit times be achieved if the piston were held down, then released when the pressure had built up? From the tests just described, it was concluded that it was very unlikely that anything could hold down the piston.

6. SEFOR SITE ACTIVITIES

6.1 The Helium Incident

At this point, early 1970, the FRED was complete and ready for the SEFOR experimental program. One incident that occurred on the SEFOR site which caused considerable concern about the FRED took place when an operator at the site charged the system with helium instead of argon. The FRED response was much faster (10%) and as tests were continued and argon reintroduced into the system the FRED response began to converge back to the earlier results. The operator was unaware that the type of gas made a difference, since no special precaution had been noted in the operating procedures although the type of gas to be used was given. This caused considerable excitement because the FRED was bing inconsistent. A number of explanations were proposed as to what was happening such as binding or friction in the system. Fortunately, the operator was honest and came forward and reported what had happened. To check things out, the incident was simulated on the analytical computer model with similar results. There was enough safety margin that no problems were encountered because of the incident.

This is one of the difficulties encountered by the designer: operator error. In ordinary design work, most of the effort is concentrated on normal operating conditions. In the case of the FRED much of the effort was concentrated on abnormal conditions which resulted in a large margin of safety.

6.2 Test Results

Before the SEFOR transient tests using the FRED were conducted, a series of static tests (referring to the static reactivity) and oscillator tests (referring to a sinusoidal oscillation of reactivity) were conducted in the range of 0 to 20 MWt of power. These tests were designed to measure the various reactor power coefficients, including the Doppler

coefficient. As a result of these tests, the Doppler coefficient of SEFOR was calculated to be -0.0081 ±0.0014 (±17.3%). This was in good agreement with the predicted value of -0.0085 ±20%.

The transient tests using the FRED were then carried out, starting with the sub-prompt critical testing using poison rod worths of 93 to 98 cents at power levels of 5, 10, and 15 MWt. The Doppler coefficient calculated from these tests was -0.0081 ±0.0011, in very close agreement with the previous test results.

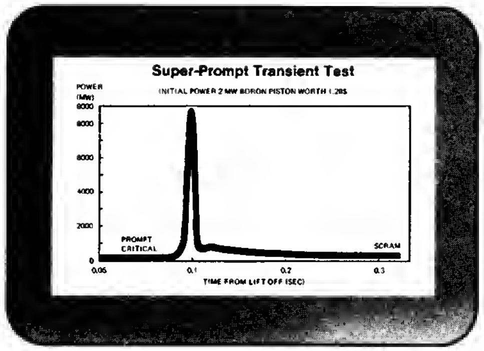
The super-prompt critical tests were finally conducted in 1970. These tests involved the rapid insertion of 1.0 to 1.3$ of reactivity by the FRED, to test the ability of the Doppler coefficient to reverse the power ascent and bring the reactor to a safe operating level. Fig. 6.1 shows the reactor power curve starting with an initial power of 2 MWt and a poison slug worth 1.28$. The power increased gradually until the reactivity reached one dollar at about 0.07 second. At this point, the power rose astronomically, reaching 9,000 MWt in a few milliseconds, at which time the Doppler effect took control and reversed the power. The reactor scram at 0.3 second after the FRED piston lift-off completed the reactor shutdown by removal of the neutron reflector around the core.

Fig. 6.2 shows the power trace for an initial power of 5 MWt and the same poison slug worth. Also included are curves for peak fuel temperature, peak fuel clad temperature, and peak sodium temperature as functions of the time after the FRED piston lift-off. The steep fuel temperature rise just before 0.1 second produced the strong negative Doppler effect that reversed the power excursion and held the reactor power at a safe level. The Doppler coefficient determined by these tests was -0.0081 ±0.0012. Not only was the Doppler coefficient reconfirmed, but the super-prompt critical tests provided a convincing demonstration that the LMFBR possesses a prompt negative Doppler coefficient, making it inherently safe and very stable.

7. EPILOG

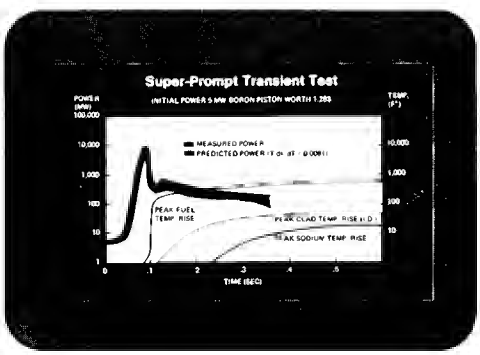
After the Doppler measurement tests, General Electric hoped to participate in the SEFOR Follow-on Program. The purpose of the Follow-on Program was to conduct fuel safety experiments, as opposed to the previous physics safety experiments. These experiments included overpower and undercooling behavior.

The FRED would be used in experiments in which the reactor would be operated at higher initial power levels, with subsequent firing of the FRED. The resulting power curves would then be



Super-Prompt Transient
Power Trace

FIGURE 6.1



Super-Prompt Transient Test Details

FIGURE 6.2

compared to the analytical results. Dr. Paul Greebler and others had predicted that the Doppler effect would diminish as the operating temperature levels increased (due to higher power levels). The ultimate goal of these experiments was to run the reactor up to core meltdown temperatures, then fire the FRED.

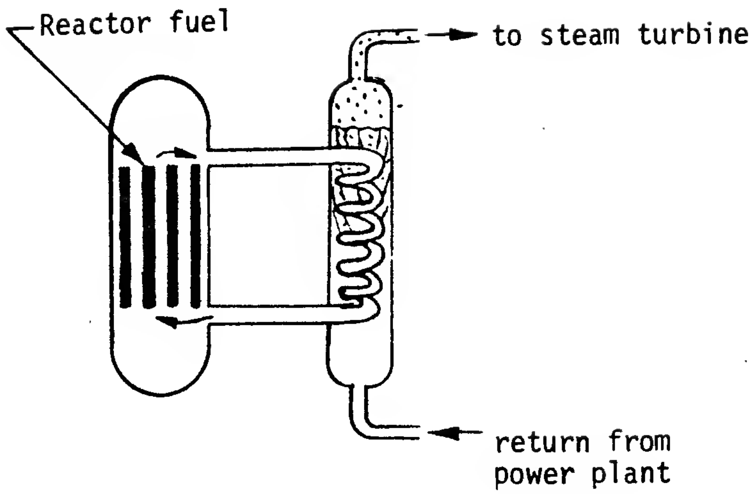
However, the funding of the Follow-on Program by the AEC was not approved, and SEFOR operations ceased in the early months of 1972. In the spring of 1972, plans to turn over the SEFOR site for educational purposes were being considered.

APPENDIX A

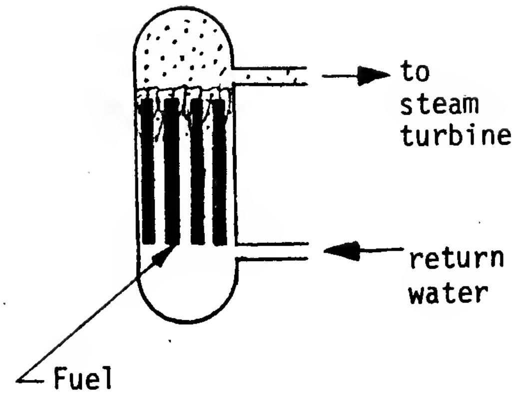
Theory of Thermal and Breeder Reactors

Nuclear reactors can be classified into two types, depending on the speed at which the neutrons are used in the reaction process. In conventional reactors, the neutrons emitted by the fissioning process are slowed by a moderator (usually water) to increase the likelihood of collision with a fertile atom, which then converts to a fissionable atom and maintains the chain reaction. Since the "slow" neutrons are in thermal equilibrium with their environment (i.e., travel near the velocity of gas molecules at ordinary temperatures), this type of reactor is often called a thermal reactor. The water moderator may be pressurized (Pressurized Water Reactor, PWR) or allowed to boil (Boiling Water Reactor, BWR), and it may be light water (ordinary water Light Water Reactor) or heavy water, D_2O, (Heavy Water Reactor). Diagrams of the PWR and BWR are presented in Figure A.1.

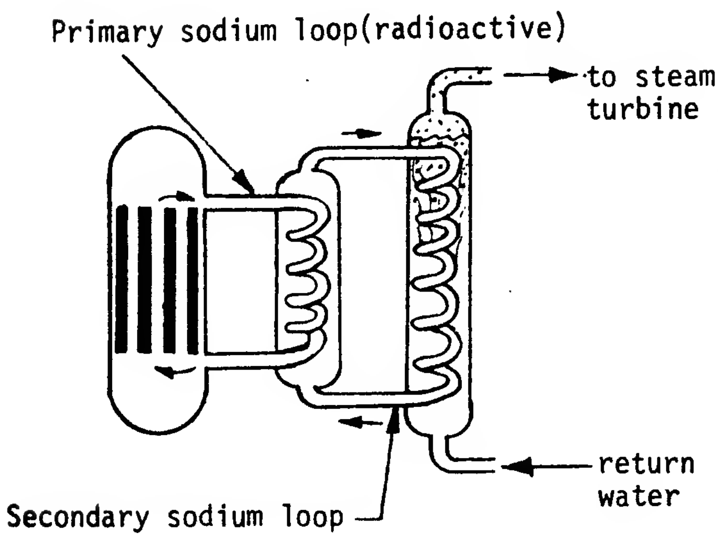
At the same time thermal reactors were suggested in the early 1940's, a second type of reactor was considered. In this type of reactor, there is no moderator and the neutrons are used at the same speed at which they are emitted from the fissioning atoms; hence it is called a fast reactor (for "fast" neutrons). The mean neutron energy is around 1 Mev or more. Although the chance of one of these fast neutrons causing fissioning is much smaller than in a thermal reactor, these reactors use a more active fuel with a higher fissioning rate. By using fast neutrons, a greater percent of the fertile material in the fuel can be converted into fissionable fuel. In fact, since each fissioning process produces over two neutrons on the average, fast reactors can actually produce more fissionable fuel than is used up. Since fast reactors "breed" additional fuel, they are called breeder reactors. The working fluid which is used to cool the core and to generate steam for the electric generators must not act as a moderator. Liquid metals such as sodium do not appreciably affect the speed of the neutrons passing through them and can be used to cool breeder reactors. The experimental breeder reactor involved in this case (SEFOR) was a sodium cooled Liquid Metal Fast Breeder Reactor. A diagram of a liquid-metal cooled breeder reactor is shown in Figure A.1. Thermal and breeder reactors may also be gas cooled, as shown in Figure A.1.



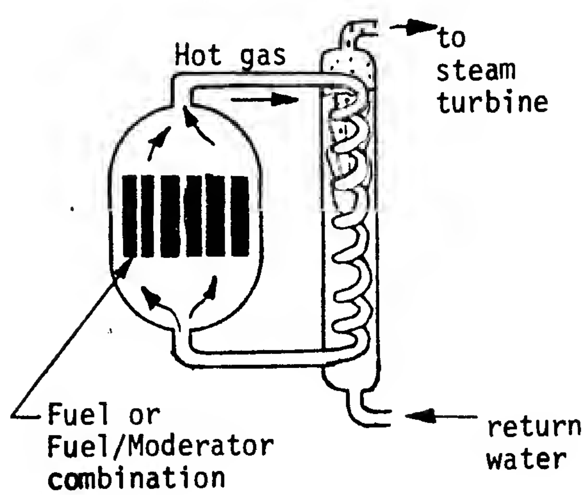
a. Pressurized Water Reactor



b. Boiling Water Reactor



c. Liguid Metal Fast Breeder Reactor



d. Gas Cooled Reactor

Figure A-1. Types of Nuclear Reactors.

APPENDIX B

Ecological Incentives for the Breeder Reactor[13]

By using a ceramic type of fuel, breeder reactors can operate at higher temperatures than conventional water reactors (1000°F or more versus 600°F for water reactors[14]). This leads to an estimated plant thermal efficiency of 43%, compared to 39% for coal fired plants and 32% for water reactor plants. This reduces the amount of thermal waste, which means less harm to the environment. Comparing wastes from power plants of 1000 MW capacity, a coal fired plant would produce over 21,000 tons of gaseous pollutants per day (over 7.6 million tons per year). A nuclear power plant would produce essentially no gaseous or liquid waste (about 5 mrem/year or less of radioactive wastes. This compares to background radiation of 140 mrem/year, and one X-ray giving 500 to 5,000 mrem). Although radioactive wastes are usually considered more hazardous than "normal" forms of air pollution, Stig O. Bergstrom of Sweden, in his paper "Environmental Consequences from the Normal Operations of an Urban Nuclear Power Plant", found that nuclear plants showed a health improvement of 10,000 times over an oil fired plant, which has sulfuric acids and other harmful pollutants.

In terms of solid wastes, a 100 MW coal fired plant would also produce 7,350,000 ft^3 of fly ash per year (330,000 tons). An equal sized water reactor plant and fuel processing plant combined would produce 5,100 ft^3/year of radioactive wastes, and an equal sized LMFBR plant would produce only 100 ft^3/year of radioactive wastes (from the processing plant only).

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13 Information obtained from "Ecological Considerations and the Fast Breeder Reactor", by A. S. Gibson (G.E.).

14 "Rebirth of the Breeder", Clare E. Wise, Machine Design, March 23, 1972.

APPENDIX C

ANALYSIS

3.1 Dynamic Analysis

A detailed analysis was carried out to determine the dynamic characteristics of the FRED for comparison with experimental test results. The configuration of the FRED used to develop the analytical model is shown in Figure 11. The following assumptions were made in carrying out the analysis:

1. The gas flow is adiabatic and reversible.
2. There is no gas leakage from the system.
3. There is no sliding friction between the piston and cylinder wall and between shaft and shaft seal.
4. The perfect gas law can be used to calculate the pressure in each chamber.
5. The work of lifting the piston is done by the gas in the low pressure chamber.
6. The gas compression in the cushion chamber is an adiabatic process.
7. Stagnation conditions occur in the expansion chamber.

With the above assumptions and the model shown in Figure 11, an analytical model was developed and then programmed for solution by digital computer. The equation of motion for this model was obtained by solving the following equation:

$$\ddot{x}_p = \frac{\dfrac{m_3 R T_0 A_3}{V_{03} + A_3 X} - \dfrac{m_4 R A_4 T_{04}}{(V_{04} - A_4 X)} \left(\dfrac{\ell - x}{\ell}\right)^{k-1} - m p g_c}{m_p \left(1 + \dfrac{R A_3 X_p}{C_p J \,(V_{03} \pm A_3 X)}\right)}$$

This equation is arrived at by taking a force balance on the piston. The assumptions used neglect nonrecoverable losses and thus tend to produce faster withdrawal times than the experimental data, which are presented in Section 5 and shown in Figure 12. One method used to compare solutions of the above equation with experimental data was by plotting the time to travel 20 in. as a function of accumulator pressure. Plots of experimental and analytical results are shown on Figure 12.

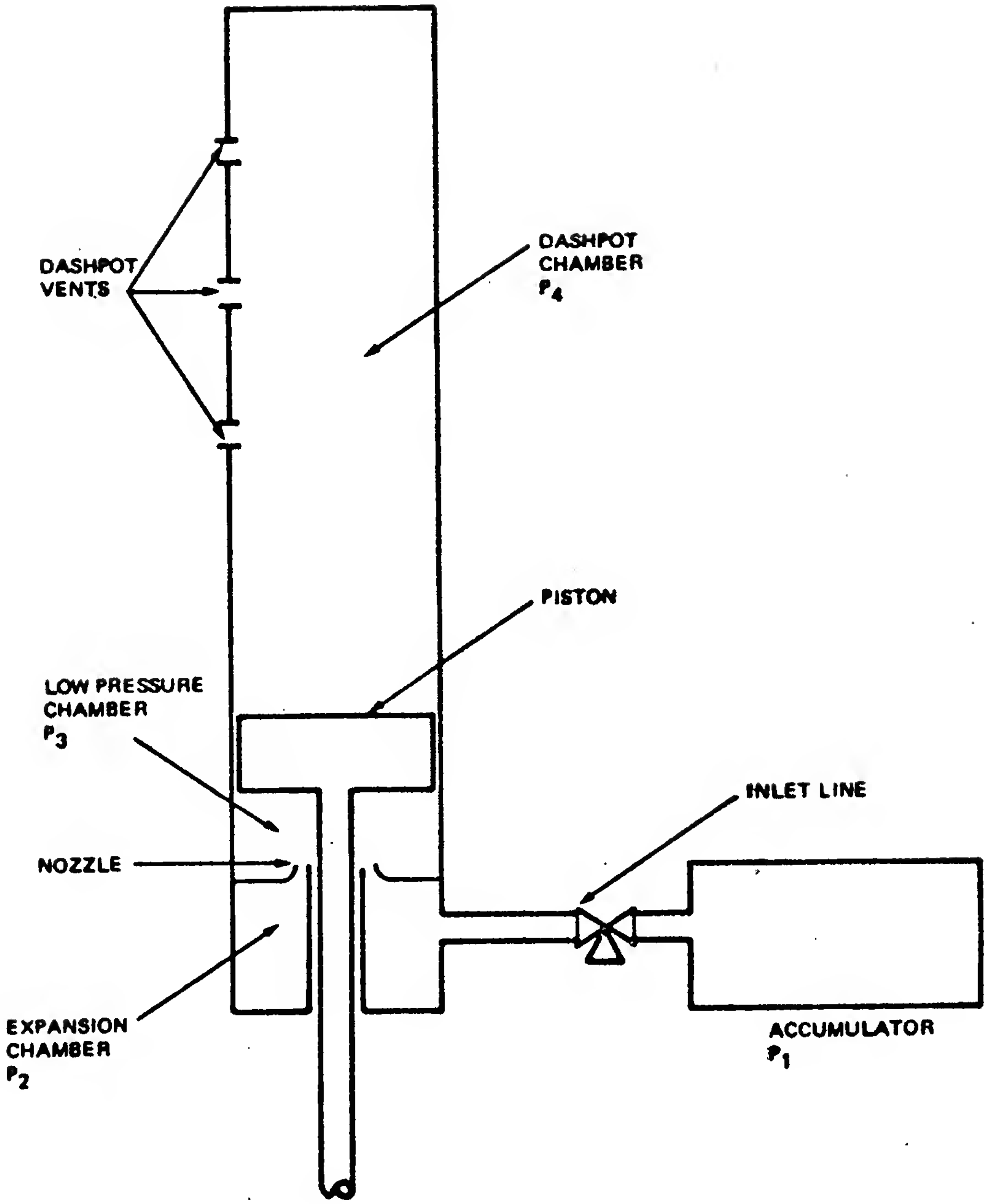

FIGURE 11 - FRED Analytical Model

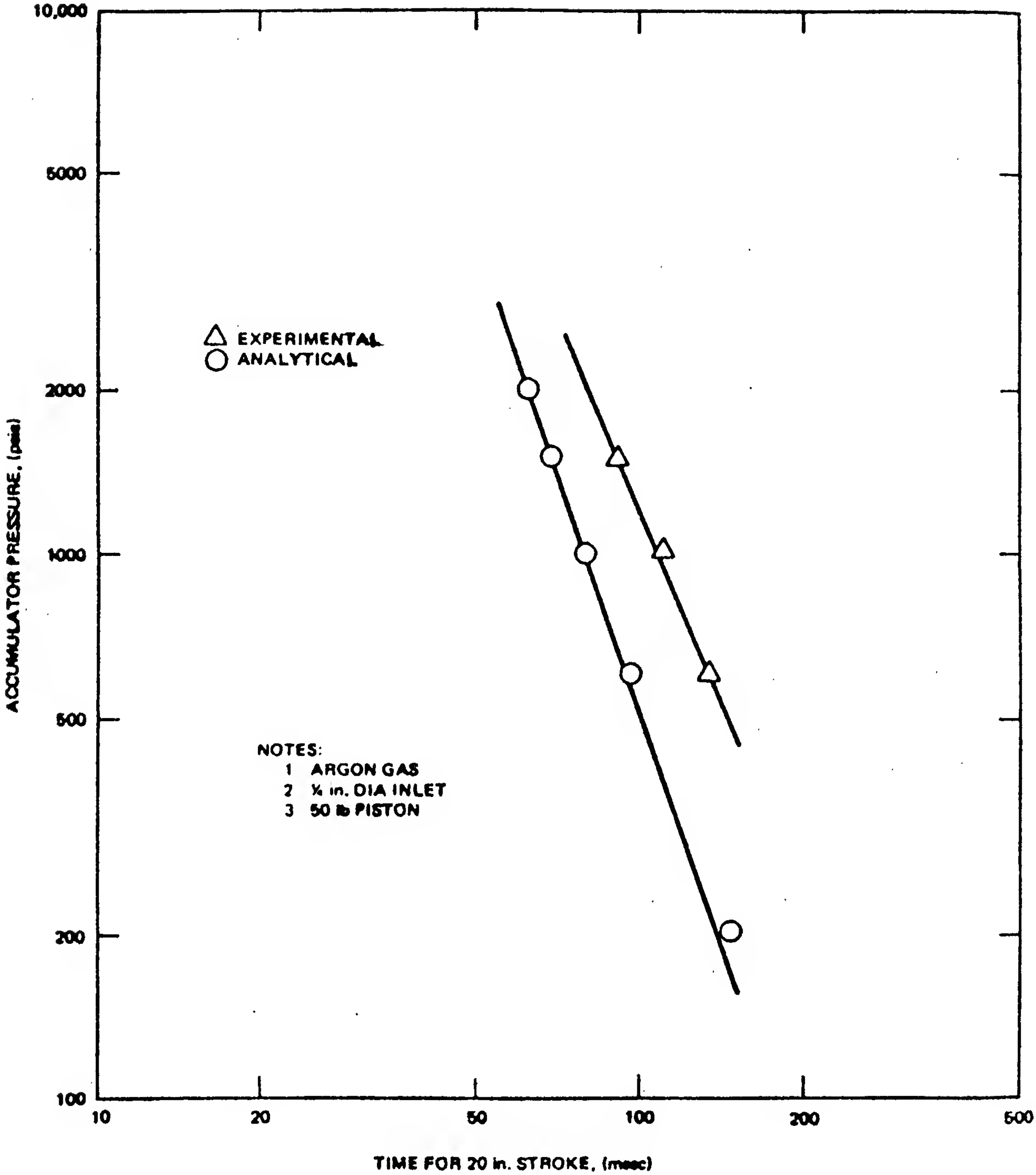

FIGURE 12 - FRED Characteristics

These lines (plots) have a characteristic equation of the form

$$t = BP_{01}^{-m}$$

where P_{01} is accumulator pressure, t is time to travel 20 in., and m and B are constants. The main difference between the experimental and analytical results is due to the effects of fluid friction, fluid loss coefficients and gas leakage, which are not included in the analytical model. If the losses were included, assuming constant coefficients, the analysis shows that the curves would be displaced in time but would be parallel to the curve as shown on Figure 12. (However, to provide a true representation of the fluid losses, the coefficients must vary with the flow rate.) It is important to note that the results of the analysis and the experimental data both show the same basic characteristics. Both show that the device has a self-limiting characteristic; they indicate that the sensitivity of the withdrawal time to increasisng accumulator pressure in the region of FRED operation is small.

Thus an increase in accumulator pressure from 1500 psi to 2000 psi changes the withdrawal time from 90 msec to slightly over 80 msec. The variations of pressures in each chamber of the analytical model are shown in Figure 5 for an accumulator pressure of 1500 psi. These show that the peak pressure in the expansion chamber is under 200 psi.

3.2 Stress Analysis

A detailed stress analysis has been carried out on the various components of FRED. The loadings and pressures used in this analysis are well above those expected during operation of FRED. FRED was fabricated from tubular shapes of 1/4 in. wall thickness or larger and thus the tensile stress due to axial loading is small in all components. In the case of the support column, the reactions due to FRED actuation creates less than 1000 psi of axial stress.

The following is a description of the loading and maximum stress in each component:

1. Accumulator - The accumulator is designed for 2000 psi at 150°F and is expected to be operated at 1500 psi and 130°F. It is fabricated of Type 304 stainless steel schedule-40 pipe sections. The maximum principal stress at the 2000 psi design pressure is 16,600 psi, which is less than one-half the yield strength of Type 304 stainless steel.

2. Support Column - The support column is the bottom piece of FRED, in which the photoelectric cells are mounted. Two types of loading occur at the lower flange.
   a. bending loads due to side loads.
   b. axial loads due to the piston reaction, or when the grapple picks up FRED, positioner, and drywell for removal from the reactor vessel head.

The maximum stress of 7800 psi occurs in the bolts due to the bending load assuming a 1/2 g side load which is more than twice the design earthquake load. The stress in the tube wall for this condition is 1090 psi. This stress is still well below the allowable yield stress. The upper flange of the support column has less loading than the lower flange, therefore, the stresses will be less.

3. Expansion Chamber - The primary loading of the expansion chamber is due to pressure build-up during the actuation of the FRED. The pressure build-up is shown in Figure 5 to have a maximum of under 200 psi. The design pressure for the expansion chamber was taken as 1100 psi with the following pessimistic assumptions. The accumulator must be overcharged to 2000 psi, instead of 1500 psi, and the piston must be seated tight enough so that no gas leaks out and thus the pressure would build or equalize at 1100 psi in the expansion chamber.

   Under these conditions, the principal stress in the lower flange would be 23,400 psi. For normal operating conditions the stress will be about 1/5 of this, or under 5000 psi. The upper flange is loaded by the cushioning gas above the piston but assuming the same pressure loading, the maximum principal stress is 23,000 psi. As in the lower flange, the normal operating pressure would yield a stress less than 1/5 of the above stress. All of the above stresses are well below the yield strength of Type 304 stainless steel.

4. Cylinder - The cylinder loading is due to pressure build-up under the piston and in the expansion chamber. The load in the expansion chamber is transmitted to the lower flange of the cylinger by the orifice and as in the expansion chamber, the pressure used to calculate the loading is five times that expected in operation. The maximum principal stress in this flange due to the 1100 psi pressure is 24,000 psi which would be less than 5000 psi for normal operating conditions. The loading in the upper flange of the cylinder is due to pressure build-up during cushioning or axial loading when removing the FRED system from the reactor vessel head with the multi-ton grapple. The largest load is

the pressure build-up during cushioning which is under 100 psi and produces an axial load of about 1250 lb., which results in a stress in the fasteners of 2650 psi.

5. Cushion Chamber - The loading in the cushion chamber is similar to that in the guide cylinder and since the flange sizes on each end of the cushion chamber and the wall thickness are the same size as the guide cylinder or larger, no additional analysis on this part is necessary.

6. Piston to Actuator Rod - The connection of the piston to the actuator rod is shown in Figure 6. The loading at this joint is largest when the piston is being accelerated at the start of the stroke. The maximum pressure under the piston as shown in Figure 5 will be under 200 psi and will give a stress in the threaded section of the actuator rod of 3000 psi or about one-tenth of the yield strength of Type 304 stainless steel. The shear stress in the piston for this load is about 1800 psi.

7. Actuator Rod to Poison Rod - The joint for the attachment of the actuator rod to the poison slug is a threaded and pinned joint. Assuming that only the threaded portion is effective and that the axial load is 100 times the slug weight as compared to 25 g's expected, the stress in the threaded joint would be 4600 psi. The other stresses in the poison slug are small due to the thick cladding (minimum clad thichness is 0.1 in.).

3.3 Poison Rod Analysis

A detailed heat transfer analysis has been carried out on the FRED poison slug. The result of this analysis is given in Figure 12. The maximum rod condition will be 1$ at 15 MW, which gives a clad temperature of 1370°F. If the emissivity of the drywell was only 0.75 instead of the 0.90 that was used in Figure 13, then the maximum clad temperature would be 1550°F. A special surface treatment was applied to the drywell in the core area to increase the emissivity to over 0.90.

The above heat transfer calculations were made assuming no natural convection in the drywell and no conduction along the length of the poison rod, and a uniform drywell temperature of 800° F. All of these assumptions tend to make the calculations conservative since the drywell should run between 700° F and 820° F and some conduction and convection along the length of the rod are to be expected.

APPENDIX D

GLOSSARY OF NUCLEAR TECHNOLOGY TERMS

CENT: The amount of reactivity due to 1 percent of the reactivity from the delayed neutrons.

CLADDING: The thin lay of metal over fuel rods which contain the fuel and fission products.

CRITICAL: The condition of a reactor which is at constant power, i.e., the chain reaction is self-sustaining. The reactivity (dk/k) = 0.

DELAYED NEUTRON: A neutron emitted during the radioactive decay of a fission fragment, which can have a half-life up to 5.5 sec. Delayed neutrons make up about 0.75% of all the neutrons produced and can be emitted for several minutes after the prompt neutrons in a nuclear reaction.

DOLLAR: A measure oa reactivity, where the reactivity in a reactor due to the delayed neutrons alone is defined as one dollar (1$).

EXCURSION: A sudden rapid increase in reactor power caused by supercriticality.

FERTILE MATERIAL: A material capable of being transformed into fissionable material when it captures a neutron and experiences radioactive decay.

FISSIONABLE MATERIAL: A material which is made up of atoms which split upon being struck by a neutron, producing two (or more) fragments, one or more neutrons, and energy.

INSERTION OF REACTIVITY: The removal of a poison slug which causes a calculated increase in reactivity.

MAXIMUM CREDIBLE ACCIDENT: The worst reactor accident which is believed to be possible in the case of a foreseeable mishap and in which all safety devices function as expected, resulting in little or no fuel meltdown and no release of radioactivity outside of the containment.

MAXIMUM HYPOTHETICAL ACCIDENT: A description of what would happen if a serious accident could occur despite all safeguards. The MHA would involve the meltdown of the core, the subsequent explosive chemical reactions, and damage to reactor components.

MODERATOR: A material used to slow down neutrons in a reactor in order to increase the likelihood of fissioning.

MULTIPLICATION FACTOR (k): The ratio of the number of neutrons in one generation to the number of neutrons in the preceding generation. When k = 1, a reactor is prompt critical.

PLUTONIUM (Pu): The man-made element produced when a fertile atom of U-238 captures a neutron, giving fissionable Pu-239. Plutonium is important as a reactor fuel.

POISON: Material that absorbs neutrons unproductively, reducing the reactivity of a reactor.

POWER COEFFICIENT: The change of reactivity with increase in power.

PROMPT-CRITICAL: The condition of a reactor which is critical (sustaining a chain reaction) because of the prompt neutrons alone, without requiring the delayed neutrons. The multiplication factor k is equal to 1 when a reactor is prompt-critical.

PROMPT NEUTRONS: Neutrons which are emitted immediately after nuclear fission, making up more than 99% of the neutrons in a reactor.

REACTIVITY: The rate of change in reactor power, expressed as dk/k (where k is the multiplication factor) and having units of dollars and cents. Hence, when a reactor is critical, the reactivity is zero.

REFLECTOR: A layer of material such a graphite, beryllium, or natural uranium which surrounds a reactor core and reflects neutrons back into the core which would otherwise escape.

SCRAM: A sudden shutdown of a reactor by inserting the control rods or removing the reflector.

SLUG: Usually refers to a reactor fuel element. In this case study, the term is used to refer to the reactor poison material in the poison rod.

SUBCRITICAL: The condition of a reactor which is decreasing in power; the multiplication factor is less than 1, and the reactivity is negative.

SUPERCRITICAL: The condition of a reactor which has an increasing rate of reaction (increasing flux); the multiplication factor k is slightly greater than 1, and the reactivity (dk/k) is positive.

SUPER PROMPT CRITICAL: The condition in which the power is increasing at an astronomical rate, and the reactivity is greater than one dollar.

TEMPERATURE COEFFICIENT OF REACTIVITY: The change in a reactor due to a change in reactor temperature. A positive coefficient means the reactivity increases with an increase in temperature. A negative temperature coefficient (reactivity decreases with an increase in temperature) will help to prevent power excursions.

URANIUM: The heaviest metal found in nature, consisting primarily of the isotopes U-235, which is the only readily fissionable nucleus occuring in appreciable quantities in nature, and U-238, which may be made into fissionable Plutonium by capturing a neutron. About 99.25% of uranium ore is U-238.

WORTH: The amount of reactivity that a poison slug absorbs.